NATIONWIDE USED FUEL INVENTORY ANALYSIS

A Thesis

by

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ABSTRACT

The goal of this research was to develop a methodology to collect inventory estimates for the analysis and characterization of used fuel in the United States. To accomplish this, the Spent Fuel Database (SFD) was created. Data was collected for the database from publicly available information on the 103 operating reactors in January 2012. Using this data, plant models were developed using ORIGEN-ARP, a point-depletion tool. The output for each reactor model included current inventory estimates for used fuel taken out of the reactor 0, 1, 3, 5, 10, and 20 years ago.

To determine the applicability of the database, a methodology was developed to analyze and compare the SFD with mass values produced using knowledge of past fuel assembly designs for general reactor classes. The methodology was centered around the idea of the "applicability range" (AR) of the database, which was defined as the degree to which a correct estimate can be made quantitatively. Pressurized Water Reactors (PWRs) were shown to have a much higher AR than Boiling Water Reactors (BWRs), and older assembly classes were shown to have a lower AR than newer classes. The fission products in the database were shown to consistently have a high AR. Berkelium and californium had low AR for all of the assembly classes, curium had low AR for BWR classes and mixed AR for PWR classes, and americium and some plutonium isotopes had low AR for BWR classes.

An assessment of the inventory estimates considered the potential radiotoxicity and heat load from these masses. The radiotoxicity by ingestion decreased by about a

factor of 10 from the newest used fuel to the oldest, and the radiotoxicity by inhalation decreased by a factor of 2. While one person could never eat or inhale a spent fuel assembly, radiotoxicity was used as a metric for the upper limit of possible harm. The heat load decreased by more than a factor of 100 over the same range of fuel assemblies. On a per assembly basis, the radiotoxicity and heat load showed similar trends, with newer PWR assemblies being the highest and BWR assemblies being the lowest in both categories. Considering these results, at a potential interim storage facility, priority should be given to the oldest BWR assemblies to reduce the radiotoxic risk and heating requirements. Also, reprocessing and transmuting is highly encouraged to reduce the radiotoxicity and heat of the waste entering storage.

Finally, to continue improving the SFD, future work should seek to quantify the magnitude of the impact of variations in AR for curium and for BWR classes. Moreover, future work should incorporate the used fuel from all the shutdown reactors into the database. Even in its current form, though, the SFD is a useful reference tool.

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NOMENCLATURE

AR	Applicability Range
B&W (or BW)	Babcock & Wilcox
BWR	Boiling Water Reactor
CE	Combustion Engineering
EIA	Energy Information Administration
GE	General Electric
GUI	Graphical User Interface
ICRP	International Commission on Radiological Protection
kW	kilowatt
MTHM	metric ton of heavy metal
MTU	metric ton of uranium
MW	Megawatt
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
Sv	Sievert
SFD	Spent Fuel Database
TVA	Tennessee Valley Authority
UREX	URanium EXtraction
U.S.	United States
W	Westinghouse

WNA	World Nuclear Association
wt.%	weight percent

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1. INTRODUCTION

Scientists and lawmakers recognized the need to deal with the growing pile of radioactive waste coming from research activities, weapons programs, and nuclear power plants as early as the 1950's. In 1956, the National Academy of Sciences recommended that highly radioactive waste be put in deep geologic disposal for safe storage.[1] Eventually, policy and lawmakers chose Yucca Mountain as the location of the future United States repository, and many millions of dollars and man-hours were invested to make this project a reality. However, during the fiscal year of 2010, the program was effectively terminated,[2] leaving the nuclear industry without a plan to deal with the used fuel accumulating at power plants. To ensure the sustainability of the industry, "this generation has an ethical responsibility to begin implementing a durable, integrated management strategy…[for] disposal of spent nuclear fuel and high-level radioactive wastes."[3]

This responsibility is shared among all nuclear power participants, including academia. However, while extensive information about U.S. reactors remains publicly available, energy utilities reserve specific details about the used fuel at their power plants as proprietary. The Spent Fuel Database (SFD) developed in this thesis contains estimates of nuclide inventories in used fuel from nuclear reactors in the United States. Its purpose is to serve as a tool for advanced fuel cycle and waste management evaluations.

The way that the database does this is by providing a step between the standard information available about the reactors and the incredibly detailed proprietary information about their spent fuel. For example, the Nuclear Regulatory Commission (NRC) provides data about when the reactors were built, what type of reactors were chosen, how much thermal power they produced, and when those power levels were allowed to increase.[4] The World Nuclear Association duplicates much of this information in their *WNA Reactor Database*.[5] Another organization that provides useful data is Nuclear News, which has published its "Annual Reference Issue" for the past 14 years.[6] However, none of these sources give even ballpark figures on specific actinide and fission product concentrations in the used fuel. Accordingly, the publicly available information was used to construct plant models, which in turn were used to calculate the estimates of the used fuel inventories that make up the SFD.

Finally, the SFD should answer fundamental questions about spent fuel in the United States. For example, how does the fission product inventory vary based on a specific reactor? Approximately how many actinides are accounted for in all of the spent fuel at all of the power plants across the country, and what is their inventory distribution? This thesis seeks to answer questions such as these through analysis of the SFD inventory estimates.

1.1. Representation Format for Used Fuel Data

Before creating the SFD, it is necessary to address what sort of information it should contain. As previously noted, the overarching purpose of the database is to provide estimates of spent fuel inventories for fuel cycle and waste management studies.

From this purpose, two questions arise: which nuclides should be included in the database, and what sort of timeline would best account for both today's spent fuel and spent fuel sitting around from years past?

To address the first question, a simple way of looking at the nuclides available is to separate them into two broad categories: actinides and fission products. Actinides are the biggest contributor to how long the spent fuel will remain radioactive above the level of natural uranium ore. This is an important comparison for waste storage evaluations because it gives a time frame of how long the engineer is responsible for the used fuel. Actinides are also the biggest potential resource, as new fuels are developed using plutonium and other fissile or fissionable nuclides. Given these two factors, it is entirely reasonable to include all of them in the database.

As for the fission products, the question becomes more complicated. Fission products are the biggest contributor to short-term radiotoxicity. Moreover, many of them have negative environmental effects, such as high solubility in water. However, hundreds of possible fission products can be present in spent fuel. To keep the database a manageable size, certain fission products must be chosen as more important and therefore worth including. The factors that separate the fission products should include their half-life, their contribution to radioactivity, and special environmental concerns.

Finally, the timeline structure needs to be chosen. The current fleet of reactors in the United States did not all start operating at the same time, nor did the Nuclear Regulatory Commission approve power uprates for all of them, let alone at the same time. The timespan covered in the SFD should account for both of these factors.

1.2. Domestic Nuclear Power Fleet

The SFD contains estimated inventories for 103 U.S. nuclear reactors. These reactors comprise the entirety of today's U.S. nuclear fleet with the exception of the Crystal River power plant. The remaining reactors fall into broad categories based on vendor type but also feature specific variations based on operating histories and power levels. However, certain power plants have consequential events in their operating histories that lead to large differences in how their estimates are treated compared to reactors with less complicated histories.

Browns Ferry, Unit 1 and Unit 3: The database does not have inventories for Browns Ferry, Unit 1's spent fuel from 5, 10, or 20 years old because the reactor was shut down from 1985 to 2007.[7] It also does not have inventories for Browns Ferry, Unit 3's spent fuel from 20 years old as a consequence of the same circumstances.[8] In 1985, the Tennessee Valley Authority (TVA) shut down all three reactors at the Browns Ferry power plant due to operational and maintenance concerns. TVA restarted Unit 2 in 1991, followed by Unit 3 in 1995. As a part of its resource planning process, TVA decided to defer its decision that year of whether or not to recover Unit 1. In 2002, it decided based on numerous studies that recovering the unit would be the best option to deal with the region's growing electricity concerns. After "4 million work hours preparing the engineering and design and more than 15 million work hours modifying, replacing, and refurbishing systems and components to ensure Browns Ferry Unit 1 [could] produce electricity safely and reliably,"[7] TVA restarted Unit 1 in May 2007.

Crystal River: The database does not include any inventories for the Crystal River power plant, which has one reactor licensed by the NRC for operation.[4] In 2009, Progress Energy shut down the reactor for maintenance and repairs.[9] During the shutdown, the company also hoped to replace the plant's old steam generators. However, during the project, the containment building cracked. The building cracked again in March 2011 as Progress Energy attempted to repair it to return the reactor to operational status. Workers then found a third crack in July 2011. By June 2012, NRC inspectors reported the building did not have any more cracks, but the future of the plant remains uncertain due to economic concerns. Given these complications, this power plant is left out of the database so that the methodology for estimating the inventories can be streamlined.

Finally, **Comanche Peak, Unit 2** started operation in 1993, and **Watts Bar, Unit 1** started operation in 1996.[4] Therefore, the database does not include an inventory estimate for the spent fuel of either of these reactors from 20 years ago.

1.3. Thesis Objectives

The major focus of this research is to develop a collection of inventory estimates for the analysis and characterization of used fuel in the United States. A set of objectives were put forth to meet this goal:

1. Used fuel data collection and classification: Collect and parameterize nuclear power plant operational data. Identify information gaps.
2. Development of a plant model to fill the information gaps: Simulate plant operational histories and characteristics. Estimate used fuel inventories.

3. Analysis of the used fuel inventories with respect to plant operational characteristics: Develop methodology. Determine the reliability of the database.
4. Assessment of the used fuel inventories from the perspective of their storage and incineration/transmutation.

This approach takes advantage of publicly available information about currently operating nuclear power plants for the collection of the desired information. The plant model were built using SCALE/ORIGEN-ARP,[10] and the resulting used fuel inventories were analyzed using MATLAB[11] and Microsoft Excel.[12]

2. INFORMATION SELECTION AND ASSESSMENT

The United States (U.S.) has a fairly open and transparent nuclear industry. While some details are protected as proprietary information, both the industry and the regulatory body in the U.S. publish reports, publicly admit mistakes, and welcome feedback from the community at large.[13, 14] Moreover, they make access to all of this as easy as possible, hosting it on their websites for the public good.

Section 2 explores the route taken to meet the first goal of this research: the collection and classification of data from these online sources. The first subsection examines the sources of information that were chosen. The second subsection looks at what information was collected and how it was organized. Finally, the third subsection presents the gaps in the standard public information and the assumptions that were made to overcome them.

2.1. Sources of Information

Two U.S. government agencies provided the bulk of the data used for this research: the Nuclear Regulatory Commission and the Energy Information Administration. These agencies share information with the public about nuclear energy trends in general and about each commercial reactor specifically. The level of detail in this information is sufficient for the creation of the Spent Fuel Database.

The Nuclear Regulatory Commission (NRC) was created by the Energy Reorganization Act of 1974 as an independent agency to regulate commercial uses of nuclear energy.[13] One of its objectives is to maintain openness within the regulatory process, which includes sharing the facts and files it collects on the NRC website. One segment of the website lists specifics such as a reactor's design type, current thermal power level, past power levels, and capacity factors since 2003.[4, 15] This information forms the general basis of this research.

The Energy Department Organization Act of 1977 established the Energy Information Administration (EIA) to analyze and distribute statistics related to all energy sources.[16] On their website, the EIA publishes articles explaining various statistics related to nuclear energy generation for the edification of the general public. Furthermore, in 1995, it published a comprehensive report at the request of the Office of Civilian Radioactive Waste Management detailing inventories and storage capacities at each nuclear power plant.[17] While the inventories were not broken down into precise compositions, the report contained information such as the level of the design type for each reactor and the number of assemblies placed in the core. Most of the key data has not changed in the past twenty years, so this EIA report was extremely helpful in formulating a clearer picture of the current nuclear reactor fleet.

2.2. Classification of Information

Given the vast amount of information available through both the NRC and the EIA, the selection of relevant data was necessary. Since the goal of the SFD is to provide estimates of used fuel inventories, the relevant data was related to the fuel inside the reactors and the reactor power histories. Specifically, the collection of information focused on these data sets:

- Reactor type and design,[4]

- Fuel type and enrichment,[17]
- Month of reactor start-up and initial power,[4]
- Subsequent power uprates,[15]
- Number of fuel assemblies in the reactor,[17]
- Mass of initial uranium per assembly,[17]
- Historical capacity factors.[18]

The location of each reactor was also recorded to give future users another method of categorizing the database later on. Initially, the data was simply funneled into spreadsheets arranged in a quick manner, such as shown in Figure 1.

	A	B	C	E	F	G	H
1	Name	Reactor Type	Thermal Output (MWth)	Operating License	Renewed License	License Expires	Reactor Vendor/Type
2	Arkansas Nuclear 1	PWR	2568	5/21/1974	6/20/2001	5/20/2034	Babcock & Wilcox Lowered Loop
3	Arkansas Nuclear 2	PWR	3026	9/1/1978	6/30/2005	7/17/2038	Combustion Engineering

(a)

	A	L	M	N	O	P	Q	R	S
1	Name of Reactor	2003 Capacity Factor	2004 Capacity Factor	2005 Capacity Factor	2006 Capacity Factor	2007 Capacity Factor	2008 Capacity Factor	2009 Capacity Factor	2010 Capacity Factor
2	Arkansas Nuclear One, Unit 1	92%	92%	78%	102%	94%	83%	99%	90%
3	Arkansas Nuclear One, Unit 2	90%	99%	91%	90%	99%	91%	90%	97%

(b)

Figure 1. Sample of original table created to (a) collect power information and to (b) collect capacity factor information.

Once collected, these datasets were then organized using Microsoft Excel. The first step of the organization process was to connect the data to the correct reactor. Even among the NRC's own documents, some reactors are not listed under consistent names. Consequently, Table I was the first organized table created.[4, 18] While most of the alternative names seem obvious, this table became valuable when going through list after list that organized reactors alphabetically.

TABLE I. Name Alternatives for Certain Reactors.

Name	**Alternative Name**
Columbia Generating Station	WNP-2
Farley 1	Joseph M. Farley Nuclear Plant, Unit 1
Farley 2	Joseph M. Farley Nuclear Plant, Unit 2
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
Ginna	R.E. Ginna Nuclear Power Plant
Harris 1	Shearon Harris Nuclear Power Plant, Unit 1
Hatch 1	Edwin I. Hatch Nuclear Plant, Unit 1
Hatch 2	Edwin I. Hatch Nuclear Plant, Unit 2
Robinson 2	H. B. Robinson Steam Electric Plant, Unit 2
Summer	Virgil C. Summer Nuclear Station, Unit 1

After verifying that the information recorded was connected to the correct unit, the next step of the organization process was to create easy-to-navigate tables so that later on, the work of modeling each reactor would move more quickly. Since even before the days of the combined license,[19] built reactors within a certain design class shared more similar characteristics than variations. Therefore, a simple classification scheme is to group the data based on design type. The table created with this in mind went one step further by calculating average values for the characteristics listed, which would be useful for later applications that are more geared toward how a certain design is performing than a certain reactor.

TABLE II. Reactor Information Classification Groups and Averaged Characteristics.

Design Class	Number of Units in Class	First Month of Operation	Initial Power (MWth)	Month of Uprate	Final Power (MWth)	Number of Assemblies
PWR						
CE 14x14	5	Oct. 1977	2560	Sep. 1988	2715	217
CE 16x16	9	Apr. 1982	3148	Mar. 2000	3295	241
W 14x14	6	Dec. 1972	1488	Feb. 2007	1664	121
W 15x15	15	Aug. 1974	2627	Nov. 1999	2710	177
W 17x17	34	Apr. 1985	3260	Dec. 2000	3384	193
BWR						
GE 7x7	8	Aug. 1971	2190.5	Aug. 2001	2426.4	724
GE 8x8*	27	Mar. 1981	2980	Dec. 2003	3263	764

*This class also has an additional uprate between start-up and the final uprate. This average middle uprate occurred in June 1992, increasing power to 3146 MWth.

Table II lists the classification categories, along with their average characteristics. The averages listed were calculated using the collected information,[4, 15, 17] which can be found in Appendix A. The design classes are shown as an abbreviation followed by a multiplication factor, denoting the size of the assembly. All of the U.S. reactors use square lattices, so “14x14” represents an assembly that has 14 fuel elements on each side of the square and so on. For the abbreviations, “CE” stands for Combustion Engineering, “GE” stands for General Electric, and “W” stands for Westinghouse.

While the characteristics listed in Table II are easily averaged within design classes, the historical capacity factors for each reactor are not. Capacity factors are site-specific and even reactor-specific, based on refueling outages, power differences, and shutdowns for other reasons. The historical capacity factors for a specific unit do not bear any relationship to the industry averages, so this set of data was listed in a separate table alphabetically by reactor without any sort of grouping. Appendix A contains a reproduction of this table.

2.3. Information Gaps

Most of the data used to create the SFD was gathered from publicly available sources.[4, 15, 17, 18] However, a couple of essential records were missing. The first is the enrichment of the uranium dioxide used in each reactor over time. The EIA report mentioned previously does include fuel enrichment in its findings, but this is one of the few things that has changed since the report was written. Since recent records of this value are proprietary, the enrichment was assumed to be 4.0 wt. % across the board rather than use old numbers for each reactor individually.

The second missing record is the standard initial amount of uranium used per assembly in each reactor over time. Like the fuel enrichment, the EIA report includes this value for 1993,[17] but the initial mass of uranium is also a feature that has changed through the years.[20] Moreover, the report lists multiple possibilities for the initial loading for each assembly type and does not indicate which mass was the standard loading and which were the unique loadings for one-time projects. Therefore, an assumption was made on which of the values the standard uranium loading was based on context clues in the report. It was also assumed that this value did not change over time.

Finally, the third missing record is the list of capacity factors before 2003. Both the EIA and the Nuclear Energy Institute publish industry average capacity factors, with values dating back to 1957 and 1971 respectively.[19, 21] However, these values do not include the capacity factors for individual reactor units. They can be used as substitutes, but for this research, it was assumed that the available data for each reactor is more representative of its history than the industry averages. The implications of this assumption will be discussed more in the next section.

3. PLANT MODELS

Using the data collected from the online sources, plants models were developed to fill in the information gaps. Section 3 describes these models and how they were created to fulfill the second goal of this work: to simulate plant operational histories and characteristics and to estimate used fuel inventories. The first subsection covers the codes used to accomplish this task. The second subsection provides the methodology used to create the plant models in the code. Finally, the third subsection explains the output from the models that was used to create the SFD.

3.1. Applied Codes

ORIGEN-ARP[10] was used to develop the plant models that converted the collected survey information discussed in Section 2 into nuclide inventory vectors representing the reactors' used fuel. The output from the plant models were then stored in Microsoft Excel[12] spreadsheets and analyzed using MATLAB.[11]

ORIGEN-ARP is a depletion analysis tool contained in Scale, a comprehensive modeling and simulation suite for nuclear safety analysis and design.[10] Instead of using rigorous, time-consuming depletion calculations coupled to reactor physics transport routines, ORIGEN-ARP performs point-depletion calculations with problem-dependent cross section libraries. The libraries are compilations of data from the Evaluated Data Nuclear File ENDF/B-VI, the Fusion Evaluated Nuclear Data Library FENDL-2.0, and the European Activation Library EAF-99.[22]

The basic, three energy group cross section libraries used by ORIGEN-ARP are generated in advance from the compiled data using transport routines and cover a number of reactor environments. For both pressurized water reactors (PWRs) and boiling water reactors (BWRs), the libraries have been generated for six ^{235}U enrichments: 1.5%, 2.0%, 3.0%, 4.0%, 5.0%, and 6.0%.[10] They are valid for average burnup values between 0.0 and 72,000 MWd/MTU. A module in the code generates cross sections for a specific problem by interpolating the given cross section libraries.

ORIGEN-ARP uses the Langrangian method (by function YLAG) to interpolate cross section libraries.[10] Implementation of the method requires the determination of a unique polynomial that will pass through the function. Equation (1) presents this process for finding a polynomial of order *n-1*, passing through *n*-points in the YLAG function:

$$Y_n(X) = \prod_{i=1}^{n}(X - X_i)\sum_{k=1}^{n}\frac{Y_k}{(X - X_k)\prod_{\substack{i=1\\ i\neq k}}^{n}(X_k - X_i)} \tag{1}$$

where X is the independent variable, Y is the dependent variable, X_i and Y_i are the coordinates of the i^{th} point on the n-points used for interpolation of the function $f(X)$, $Y_n(x)$ is the interpolated value of X, and n is the number of points. The independent variables selected by ORIGEN-ARP for uranium-based fuels are enrichment, burnup, and water density.

MATLAB is a programming language and environment for numerical computation, simulation, and visualization.[11] One of its core functions is its ability to store large amounts of data in matrices and to manipulate that data quickly. It can import and export data from text files or Excel[12] spreadsheets. Given the combination of these factors, MATLAB is a useful tool to analyze the spent fuel inventories.

3.2. Methodology

To create a database containing spent fuel estimates for 103 reactors, including both PWRs and BWRs with different power histories, an integrated methodology was developed to streamline the process of generating the ORIGEN-ARP model inputs for each reactor. The input files were created using the OrigenArp Graphical User Interface (GUI)[10] included in the Scale software package, and the method utilized various assumptions to reduce the complexity of the calculations. Figure 2 presents a broad picture of the steps included in the integrated methodology, and Table III explains each step in detail.

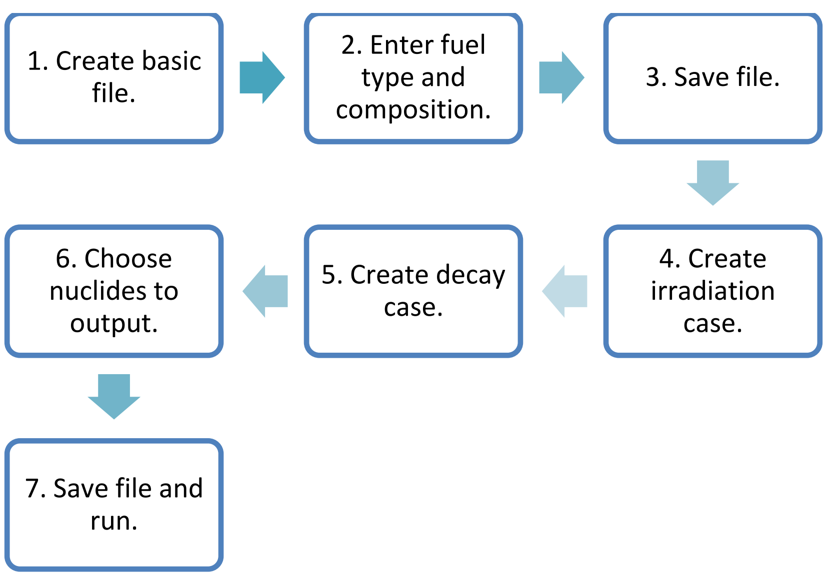


Figure 2. Steps of the integrated methodology to calculate used fuel inventory vectors.

TABLE III. Detailed explanation of the steps included in the integrated methodology.

Step	Details
1. Create basic file.	Create the basic OrigenArp GUI file using the “new” button.
2. Enter fuel type and composition.	In the “Comps” section, chose the fuel type based on the reactor classification category. (See Appendix A.) Enter the composition in grams for a total mass equal to the initial mass of uranium in the assembly, with 4.0 wt. % ^{235}U and 96.0 wt. % ^{238}U.
3. Save file.	For this work, the unique file names were composed of the power plant name, the reactor unit number, and time information if needed.
4. Create irradiation case.	Chose “Days” as the time unit. In the first empty box, enter the power calculated by Eq. (2). $$Power\ \left(\frac{MW}{basis}\right) = \frac{Reactor\ Thermal\ Power\ (MW)}{Number\ of\ assemblies} \quad (2)$$ The number of assemblies for each unit can be found in Appendix A. In the second empty box on the same row, enter the cumulative time of irradiation calculated by Eq. (3). $$Time\ (days) = 3(547.5\ days) + t_{sd,1} + t_{sd,2} \quad (3)$$ where $t_{sd,1}$ and $t_{sd,2}$ are the estimated time periods of the reactor shutdown, calculated using Eq. (4) and Eq. (5) respectively. *Continued on the next page.*

TABLE III. Continued.

Step	Details
Create irradiation case (continued).	$t_{sd,1} = \{(1 - CF_{2008}) + (1 - CF_{2009})\} * 365\ days$ (4) $t_{sd,2} = (1 - CF_{2010}) * 365\ days$ (5) where CF_{year} is the capacity factor for the reactor during the year denoted by the subscript. (See Appendix A.) Click the "Fill" button for OrigenArp to automatically separate the irradiation case into 20 timesteps. Verify that the new time steps match the outage estimates. The simplest method using 20 timesteps is to divide each irradiation cycle into six periods of 91.25 days each, then follow each irradiation with the time calculated by either Eq. (4) or Eq. (5). The start of the shutdown period is indicated by replacing the power at 547.5 days (or the equivalent time) with 0 MW/basis. Click "Ok" to exit the irradiation case, and save the file.
5. Create decay case.	Choose "Years" as the time unit. In the first empty box under "Cumulative Time," enter the oldest year at which the power used in the irradiation case was the rated power of the reactor unit. (See Appendix A.) Click "Fill," and verify that each of the selected years for output is included in the time steps given, up to the oldest year entered. Check the "Save Results" box for each of the selected years. Click "Ok" to exit the decay case, and save the file. *Continued on the next page.*

TABLE III. Continued.

Step	Details
6. Choose nuclides to output.	In the "Plot Setup" section, choose the nuclides of interest for easy access in the output file. Make sure to choose the correct output units, and request plots for both the decay case and the second irradiation case to receive nuclide concentrations for 0 years up to the oldest year included in the decay case.
7. Save file and run.	If the reactor unit previously had a power uprate, resave the file using another name, and repeat steps 4 through 7. Only change the power level in the irradiation case and the years included in the decay case. If not, move on to the next reactor.

This method includes a number of assumptions. Even before the implementation of the steps listed in Table III, the data used to generate the models were based on the assumptions listed in Section 2. Namely, the enrichment of all the fuel was assumed to be 4.0 wt. % ^{235}U, and the initial mass of the uranium in each assembly was assumed to be the most likely value given for 1993.[17] For both of the factors, the assumption was made that these numbers were valid across the timespan considered for the database.

For the method presented in Table III, Equation (2) is based on the assumption that the reactor operated at 100% power whenever it was not shut down for refueling. While this would greatly affect the day-to-day composition of the fuel in the reactor, after months of operation, the fuel should reach an equilibrium composition similar to that of fuel in a reactor that was continuously operated. Therefore, this effect would be diminished and should not produce a noticeable change in the database inventories.

Equations (3) through (5) are based on the assumption that, on average, the reactors shutdowns every 1.5 years (or 547.5 days) for refueling and that only one-third of the reactor core is refueled at any one time. Therefore, one assembly would undergo three cycles of irradiation. It is also assumed that all downtime and maintenance occurs during the refueling outage, or in other words, one shutdown period follows every cycle, which led to Eq. (4) and Eq. (5). The database uses January 2012 as the current reference date, which is why the capacity factors for 2008 and 2009 were chosen for the first shutdown (assuming the outage occurred around end of the year) and why the capacity factor for 2010 was chosen for the second shutdown. It is important to include the shutdown periods in the irradiation case to account for the decay of fission products during zero power.

Finally, the assumption was made that the available capacity factor data for each reactor is more representative of its history than industry averages. The outcome of this assumption is that the capacity factors for 2008 to 2010 were used to represent the capacity factors for all of the scenarios for each reactor. Therefore, the model underestimates the shutdown periods for the older time periods. However, since the database only includes fuel that has been stored for up to 20 years, the difference in the spent fuel inventories was assumed to be negligible.

3.3. Model Output

Having selected the correct nuclides in step 6 of the methodology, the output file created by ORIGEN-ARP contains tables of the nuclides' mass at the selected decay time periods. The decay periods chosen for the SFD are listed in Table IV. Even though the database neglects the oldest spent fuel, the chosen decay time periods are ideal for this research. They account for the hotter spent fuel assemblies, which would be of more concern in the event of a widespread disaster, and the time periods considered are short enough that the assumptions made to simplify the methodology remain valid.

TABLE IV. Decay time periods modeled in ORIGEN-ARP and their associated reference dates.

Decay Time (years)	Reference Date of Placement into Storage
0	January 2012
1	January 2011
3	January 2009
5	January 2007
10	January 2002
20	January 1992

The nuclides chosen for inclusion in the SFD are a mixture of actinides and fission products, shown in TABLE V. For the actinides, all of the nuclides were included in the database for each element from thorium to californium. These nuclides are especially important for waste storage studies since they produce the bulk of the long-term radioactivity. They are also potentially useful for other purposes, such as studies aimed at reprocessing, since the separation of useful material from pure waste generally focuses on actinide removal.

TABLE V. Actinides and fission products contained in the SFD.

Nuclides found in the SFD						
Am-239	Cf-249	Cm-248	Np-236	Pd-107	Sb-125	Th-234
Am-240	Cf-250	Cm-249	Np-236m	Pm-147	Se-79	U-230
Am-241	Cf-251	Cm-250	Np-237	Pu-236	Sm-151	U-231
Am-242	Cf-252	Cm-251	Np-238	Pu-237	Sn-126	U-232
Am-242m	Cf-253	Cs-134	Np-239	Pu-238	Sr-90	U-233
Am-243	Cf-254	Cs-135	Np-240	Pu-239	Tc-99	U-234
Am-244	Cf-255	Cs-137	Np-240m	Pu-240	Th-226	U-235
Am-244m	Cm-241	Eu-154	Np-241	Pu-241	Th-227	U-236
Am-245	Cm-242	Eu-155	Pa-231	Pu-242	Th-228	U-237
Am-246	Cm-243	He-4	Pa-232	Pu-243	Th-229	U-238
Bk-249	Cm-244	I-129	Pa-233	Pu-244	Th-230	U-239
Bk-250	Cm-245	Kr-85	Pa-234	Pu-245	Th-231	U-240
Bk-251	Cm-246	Nb-94	Pa-234m	Pu-246	Th-232	U-241
Cd-113m	Cm-247	Np-235	Pa-235	Ru-106	Th-233	Zr-93

The fission product nuclides contained in the SFD were chosen more selectively. The long-lived fission products with half-lives of over 25 years were chosen because they must be considered in any study aimed at decreasing the radiotoxicity of spent fuel in storage.[23] They are ^{135}Cs, ^{137}Cs, ^{129}I, ^{94}Nb, ^{107}Pd, ^{79}Se, ^{151}Sm, ^{126}Sn, ^{90}Sr, ^{99}Tc, and ^{93}Zr. Fission product with half-lives longer than one year were also chosen if their contribution to radioactivity is non-negligible. These nuclides are ^{113m}Cd, ^{134}Cs, ^{154}Eu, ^{155}Eu, ^{85}Kr, ^{147}Pm, and ^{125}Sb. Additionally, the database contains concentrations for ^{106}Ru and ^{4}He. It is important to note that ORIGEN-ARP did not restrict itself to using only these fission products in the point-depletion calculations; rather, these were the only ones selected to be included in the easily accessible tables.

With these selected characteristics, the SFD was compiled in Microsoft Excel. The tables in the ORIGEN-ARP output files were copied directly into separate spreadsheets, and the spreadsheets were organized by reactor unit.

4. ANALYSIS OF DATABASE APPLICABILITY RANGE

As discussed in Section 3, the models used to create the SFD were generalized versions of their real-life counterparts. They included certain assumptions, and any variability in the data could distort the results stored in the database. Of the assumptions listed in Section 3, two critical assumptions were those regarding enrichment and initial uranium content. The ^{235}U enrichment was assumed to be 4.0 wt.% for all of the reactors, and the initial uranium content was chosen for each reactor based on context clues. These two metrics are chosen for this analysis because they affect the composition of the spent fuel most directly and because a great deal of information is available detailing their variability.

A new method was developed to quantify how the variability of these two metrics could affect the reliability of the SFD. A traditional sensitivity analysis could not be performed on the mass metrics alone because the sample size within the database was too small and the choices of enrichment and of initial uranium content were not random choices. The levels for both were decided based on what was best commercially for each reactor according to specific utility practices. Therefore, the distributions of all the enrichment levels and of all the uranium content levels used by industry over the past forty-some years are not normal distributions. Within the field of statistics, certain analysis methods are available to understand non-normal distributions, but they are beyond the scope of this work. Instead, a simple methodology was developed to assess what will be called the “applicability range” of the database.

4.1. Applicability Range

For the purpose of this research, the ***applicability range*** (AR) is the degree to which a correct estimate can be made quantitatively. It is defined as a way to simply and numerically relate the accuracy of the estimated masses listed in the SFD. The AR is a measure of how sensitive the database is to possible variations in the enrichment and initial uranium content, outside of the traditional statistical definition of "sensitivity."

More specifically, AR is a way to classify the percent difference between the mass listed in the database and the mass that would actually be present given a certain enrichment and initial uranium content of a fuel assembly. In the analysis to follow, the AR is split into three classifications:

- High AR (0% to 10% difference),
- Moderate AR (10% to 25% difference),
- Low AR (more than 25% difference).

The "high AR" classification groups together those nuclides whose value does not change very much with variation in enrichment and initial uranium content. Similarly, the "moderate AR" classification groups together those whose value does change but not so much as to completely distrust the mass listed in the database. Numerically, this was chosen to be from 10.0% to 25.0% difference. Beyond 25.0% difference with the mass listed in the database, the nuclide would fall into the "low AR" classification. A specific way to "score" each nuclide within this classification system will be discussed in subsection 4.3.

4.2. Pre-analysis Investigation: Broad Changes Caused by Variation in Enrichment

Before developing the analysis methodology for AR, an initial assessment was made to determine the necessity of such an analysis. The initial assessment looked at how the used fuel composition could change based on variation in enrichment alone. Based on the large variation over the years, a simple valuation was made to gauge this necessity.

Figure 3 was compiled based on publicly available data[20] and shows the frequency of different enrichment levels used by industry up until 2004. The histogram shows the variation in enrichment from 1.5 to 5.0 wt.% enrichment in 0.05% increments, although the bin for 1.5 wt.% is slightly misleading in that it includes all of the reactors below that value. As displayed in Figure 3, the spread around the assumed value of 4.0 wt.% is quite large, with 92.8% of the assemblies falling below that value. From this histogram alone, variation in enrichment must have some impact on the database, since the older, lower enrichment assemblies have such high frequencies, and the newer, higher enrichment assemblies are far fewer.

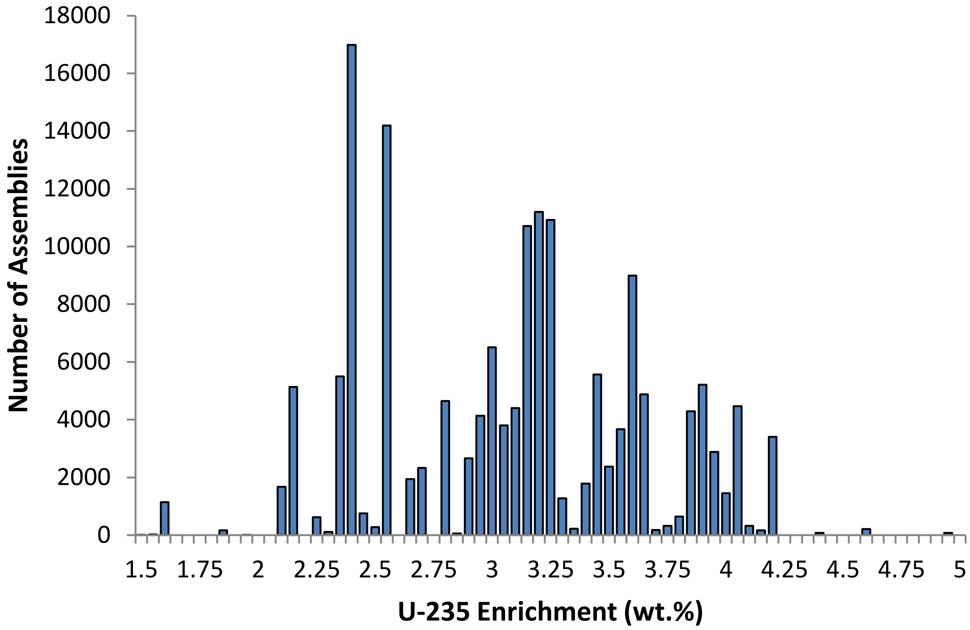


Figure 3. Histogram showing the variation of ^{235}U enrichment in fresh fuel assemblies used in U.S. reactors up until 2004.

The initial assessment used a simple methodology to numerically show possible variations in the mass estimates. First, all of the reactor models were rerun 37 times at different levels between 1.50 wt.% and 5.0 wt.% enrichment. A complete list of the enrichments used can be found in Appendix B. Then, for each enrichment level, the mass of each of the 98 nuclides in the database was averaged over all of the reactor model results. Finally, the averaged masses of the nuclides at each enrichment level were compared to the averaged masses of the nuclides at 4.0 wt.% using Eq. (2).

$$Percent\ change = \frac{mass_i^x - mass_i^{4.0\%}}{mass_i^{4.0\%}} \qquad (2)$$

where $mass_i^x$ indicates the averaged mass of a certain nuclide i in the used fuel assembly at some enrichment level x, and $mass_i^{4.0\%}$ indicates the averaged mass of that same nuclide at 4.0 wt.% enrichment.

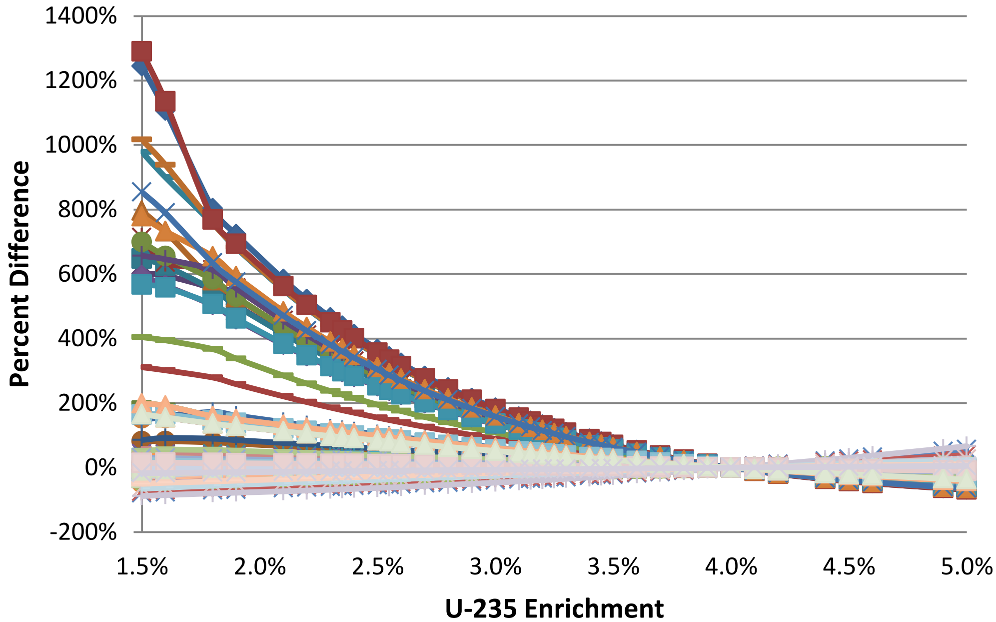


Figure 4. Percent difference from mass at 4.0 wt.% enrichment, showing all 98 nuclides in the Spent Fuel Database.

Figure 4 presents the results of this investigation. It shows the percent difference for all 98 nuclides in the database over the modeled enrichment range and is presented without a legend because the purpose of Figure 4 is to relate possible variations in the database as a whole. At 5.0 wt.% enrichment, the largest spread is from -64% to 67% difference, and at 1.5 wt.% enrichment, the largest spread is from -88% to 1290% difference. Figure 5 presents the same information as Figure 4, with the limitation that only those nuclides that were less than 10% absolute difference over the entire enrichment range are shown. Of the 98 nuclides in the database, only nine met that criterion.

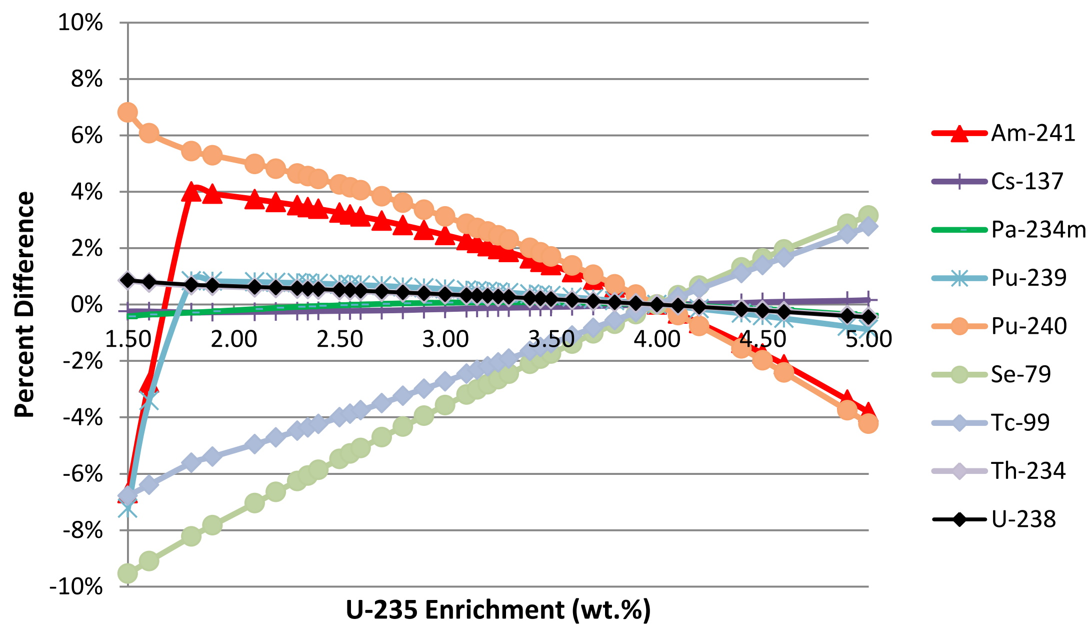


Figure 5. Percent difference from mass at 4.0 wt.% enrichment, showing only those nuclides with less than 10% absolute difference.

Together, these two figures make a strong argument for completing an analysis of the AR of the SFD. If the variation could be as large as is shown in Figure 4, and if the number of reliable nuclides could be as small as shown in Figure 5, then it is necessary to understand just how reliable the database really is. This initial assessment creates a reference point for the AR analysis. The nine nuclides shown in Figure 5 should fall into the “high AR” classification once the analysis was completed. Moreover, the hypothesis was proposed that even more nuclides would join the category once the initial uranium content was also considered.

4.3. Method for Predictability Analysis

The methodology to determine the AR of the SFD was developed based on lessons learned during the initial assessment. Furthermore, with the consideration of initial uranium content added into the mix, the system of modeling and analysis increased in complexity. This subsection presents the new method of analysis.

With the incorporation of the initial uranium content, the reactor models needed to be separated into groups. The biggest reason for this was to gain the ability to distinguish between PWRs and BWRs and between older reactors and newer reactors. Each type of reactor carries with it certain characteristics, and it would be useful to be able to determine if the database has a higher AR for one design over another.

Another great advantage of separating the reactor models into groups is that information about the assembly designs historically used in each group is publicly available.[20] Consequently, instead of varying the values blindly over a certain range, specific enrichment and initial mass values can be used to represent specific assembly designs. The reactor groups are called “assembly classes” and are very similar to the ones discussed in Section 2, which were based on the designs chosen within the ORIGEN-ARP GUI. The only change is the addition of a class for the B&W 15x15 reactors, which were included in the W 15x15 group in Section 2. Four reactor models were also separated from the rest of the reactors under this new system, as they are classified as Individual Assembly Classes: Fort Calhoun, Palisades, Saint Lucie unit 2, and South Texas. Tables found in Appendix B detail the assembly designs used by the assembly class and list which reactors belong to each class.

With the more specific numbers available for the modeling portion of the new methodology, the lessons learned from the initial investigation can be applied to the analysis. Many of the nuclides shown in Figure 4 differ from the database by much more than 100%. Therefore, instead of simply looking at the percent difference for each nuclide, a scoring system was created to emphasize those nuclides with a high AR.

Equation (3) presents the calculation of an “individual score”, which represents the database’s ability to accurately estimate masses for a specific assembly design.

$$Individual\ Score = exp\left(\frac{-ABS\left(mass_i^x - mass_i^{4.0\%}\right)}{mass_i^{4.0\%}}\right) \quad (3)$$

where ABS represents the absolute value of the quantity within the parentheses, and x now represents a specific assembly design. The fortunate aspect of this equation is that it scores everything from approaching 0 to 1.0. The transformation of Eq. (2) also has the added benefit of putting nuclides with a higher AR closer to 1.0. Equation (3) also removes directional information about the percent difference, showing only absolute fluctuations because the AR analysis is concerned with how different the mass can be and does not seek to find correcting factors. According to the classification scheme discussed in subsection 4.1, the new categories would fall from about 0.90 to 1.0 for the high AR, 0.78 to 0.90 for the moderate, and 0.0 to 0.78 for the low category.

Another lesson learned from the initial assessment was that close to 1.5 wt.% enrichment, the percent difference became quite dramatic. Instead of simply taking this huge difference to be just as important as one of the smaller differences, or vice versa, it is important to factor in how frequently that type of enrichment was used. Fortunately, part of the publicly available data on the assembly designs is how many have been used over time. Using this knowledge, Eq. (4) combines the individual scores into a weighted average to calculate an overall score for a nuclide within a particular assembly class.

$$Overall\ Score = \sum_{j=1}^{N} \left(\frac{m_j}{M}\right) (Individual\ Score)_j \tag{4}$$

where j represents a specific assembly design, N represents the total number of assembly designs used by the assembly class, m_j represents the number of assemblies used for design j up until 2004, and M represents the total number of assemblies used by that assembly class up until 2004.

Combining these ideas, a methodology was developed to determine the AR of the database. The steps of this method are organized in the flowchart shown in Figure 6. The results of the analysis will be discussed in the next subsection.

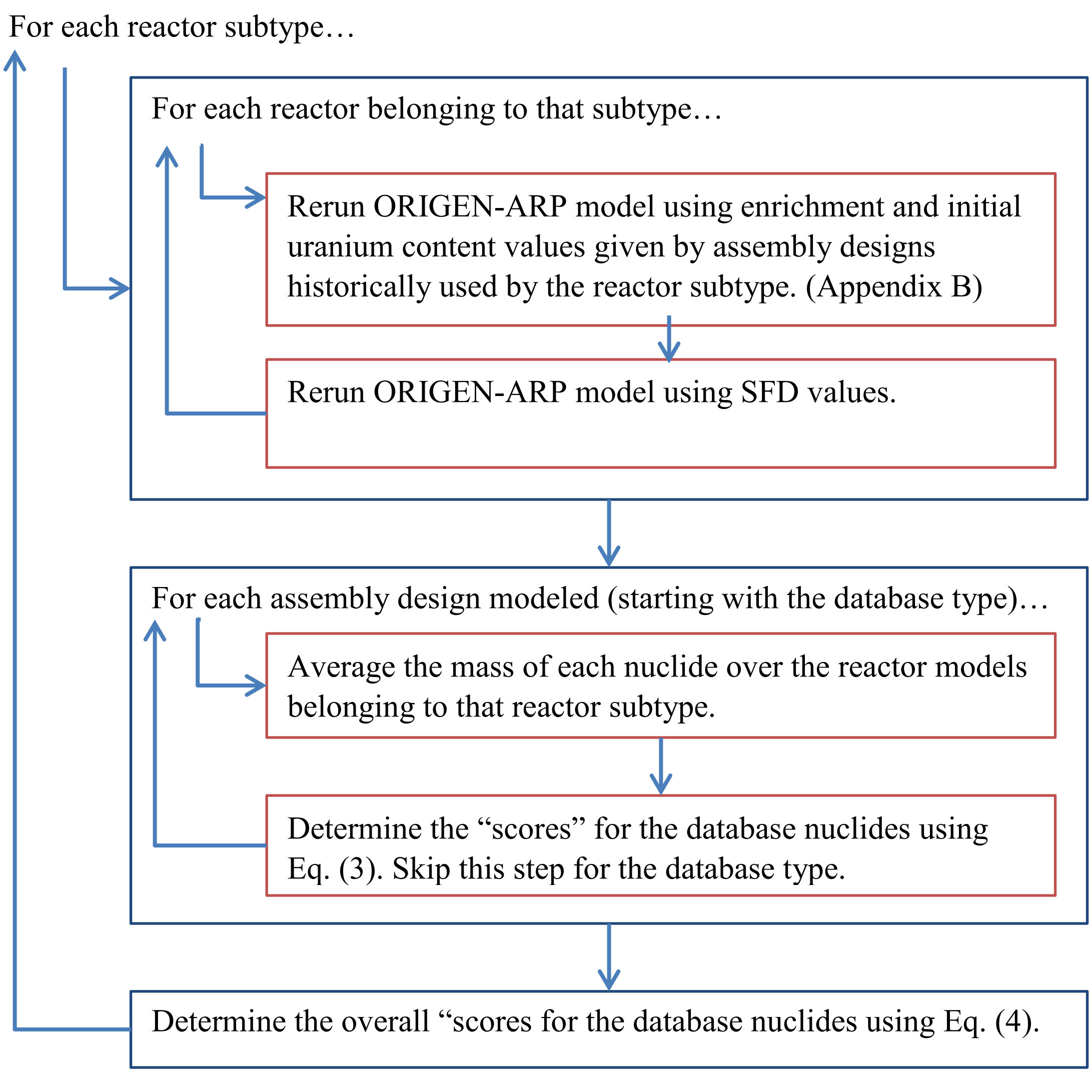


Figure 6. Flowchart describing the analysis methodology to determine applicability.

4.4. Applicability of Database for General and Individual Models

Using the method laid out in Figure 6, overall scores were compiled for the nuclides contained in the SFD for each of the assembly classes. These scores were then grouped into the three AR classifications and plotted on histograms to show how the AR fluctuates in general from assembly class to class. Figure 7 through Figure 15 present these histograms for the group assembly classes. Figure 7 presents the histogram for the B&W 15x15 assembly class; Figure 8, Figure 9, and Figure 10 present the CE assembly classes; Figure 11 and Figure 12 present the GE BWR assembly classes; and Figure 13, Figure 14, and Figure 15 present the W assembly classes. Figure 16 presents the four individual assembly classes. All of the histograms feature three shades of color to visually distinguish between the AR classifications.

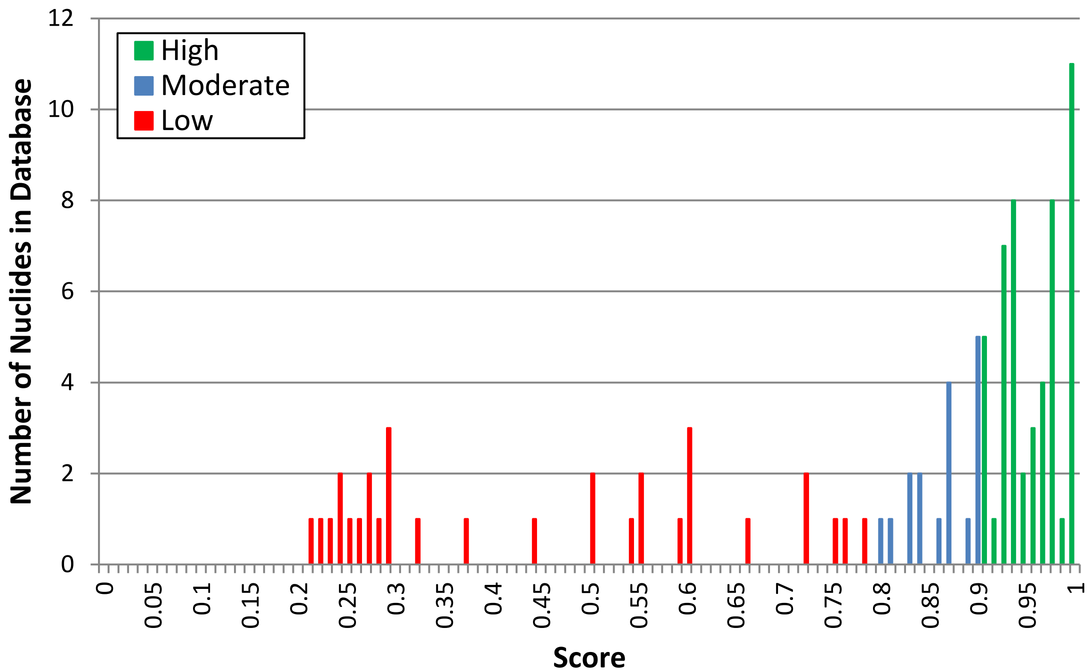


Figure 7. Overall scores for the 98 database nuclides within the B&W 15x15 Class.

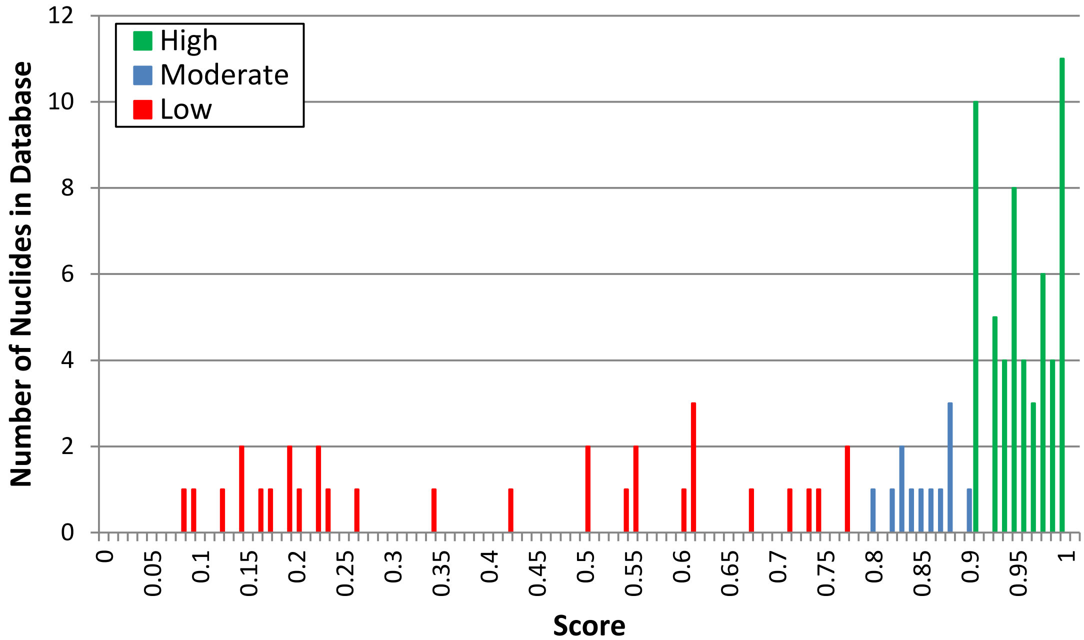


Figure 8. Overall scores for the 98 database nuclides within the CE 14x14 Class.

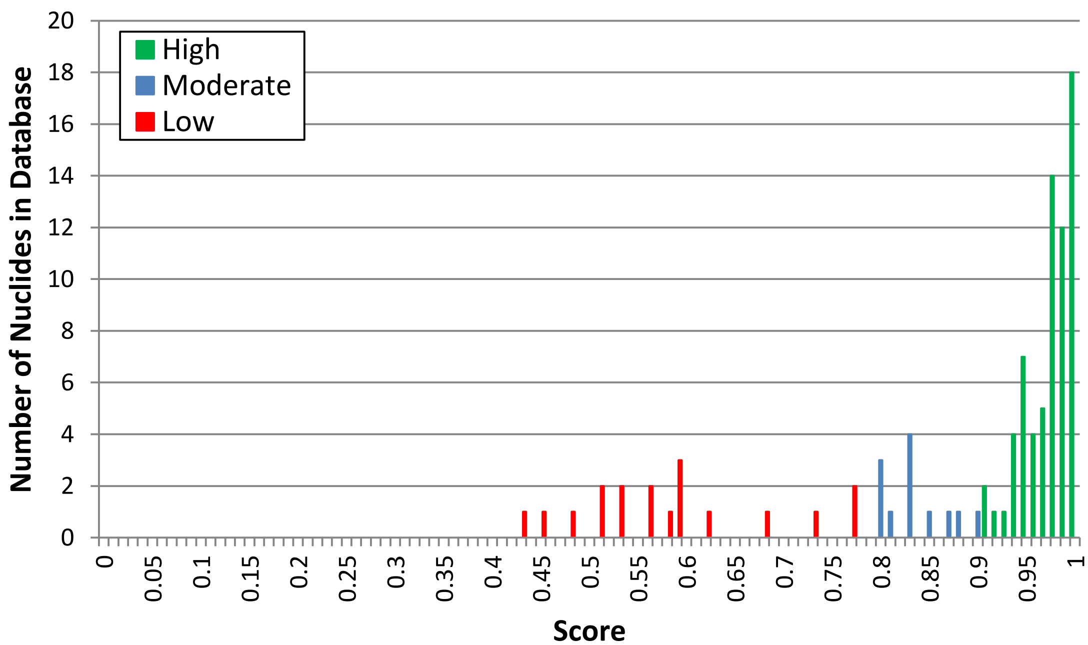


Figure 9. Overall scores for the 98 database nuclides within the CE 16x16 Class.

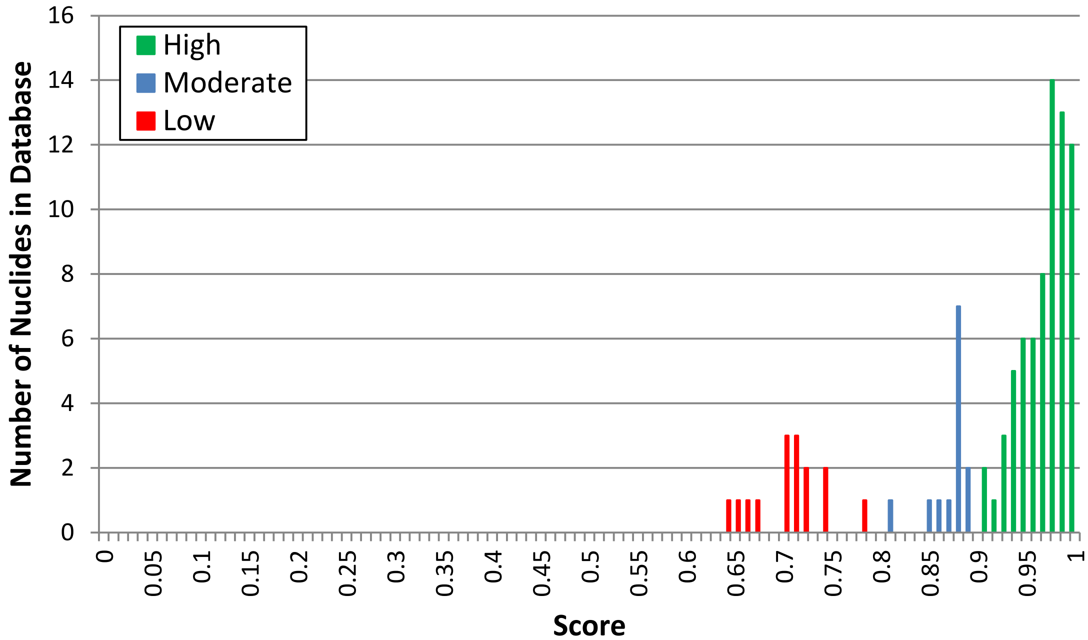


Figure 10. Overall scores for the 98 database nuclides in the CE 16x16 System 80 Class.

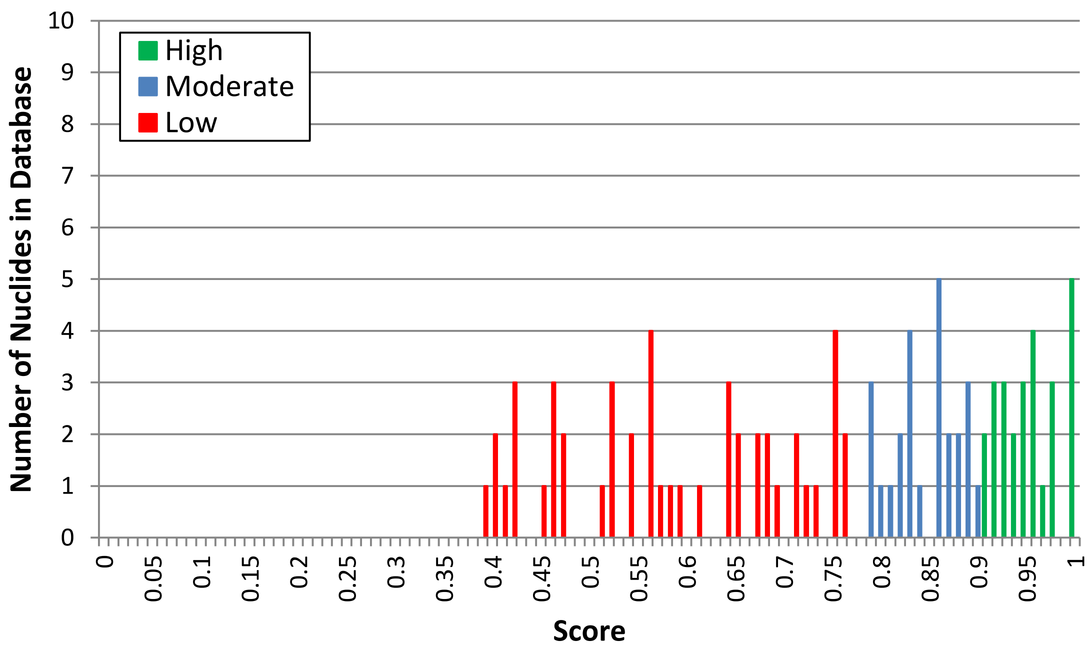


Figure 11. Overall scores for the 98 database nuclides within the GE BWR 2,3 Class.

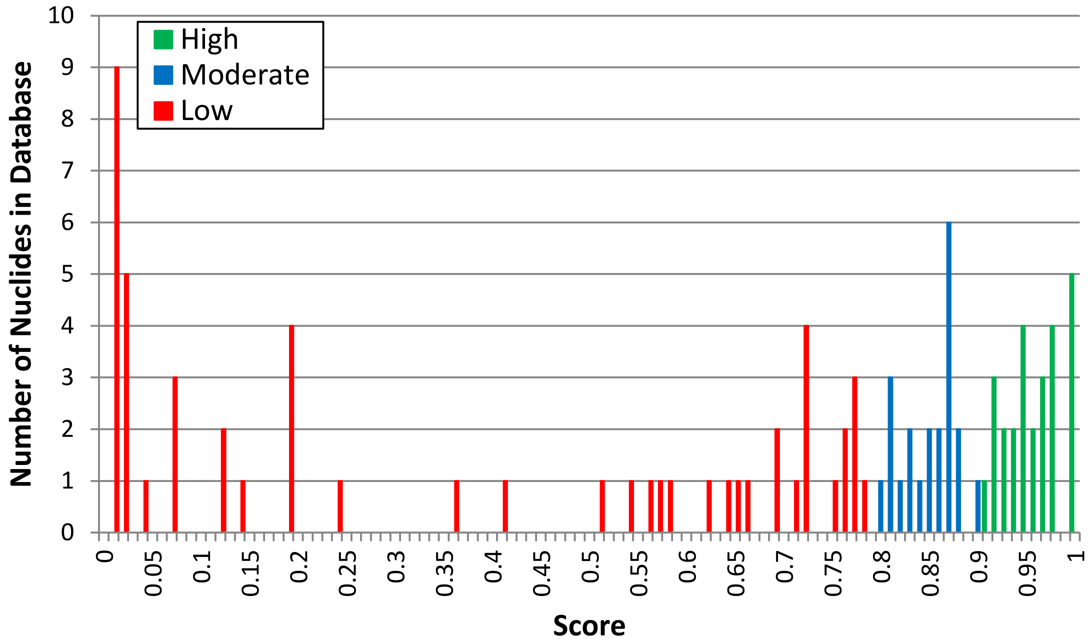


Figure 12. Overall scores for the 98 database nuclides within the GE BWR 4,6 Class.

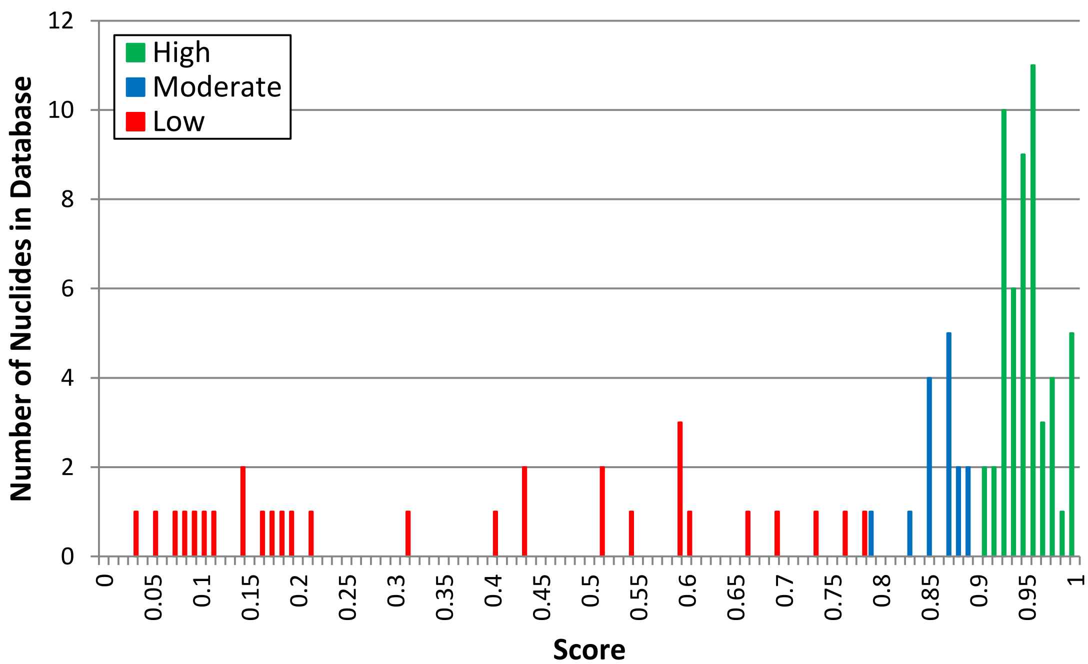


Figure 13. Overall scores for the 98 database nuclides within the W 14x14 Class.

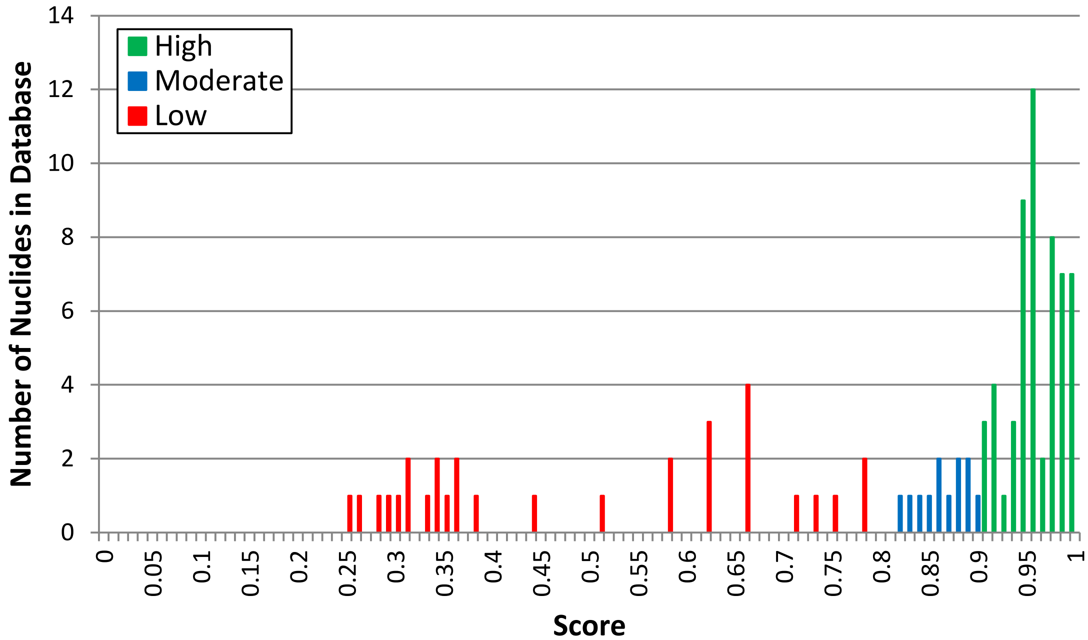


Figure 14. Overall scores for the 98 database nuclides within the W 15x15 Class.

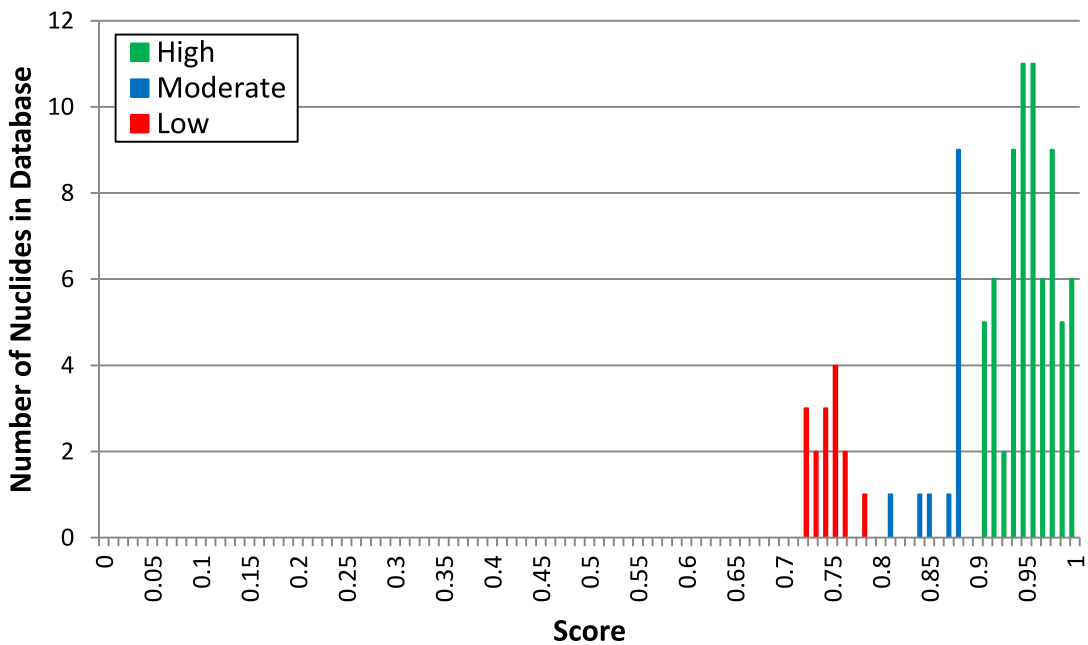


Figure 15. Overall scores for the 98 database nuclides within the W 17x17 Class.

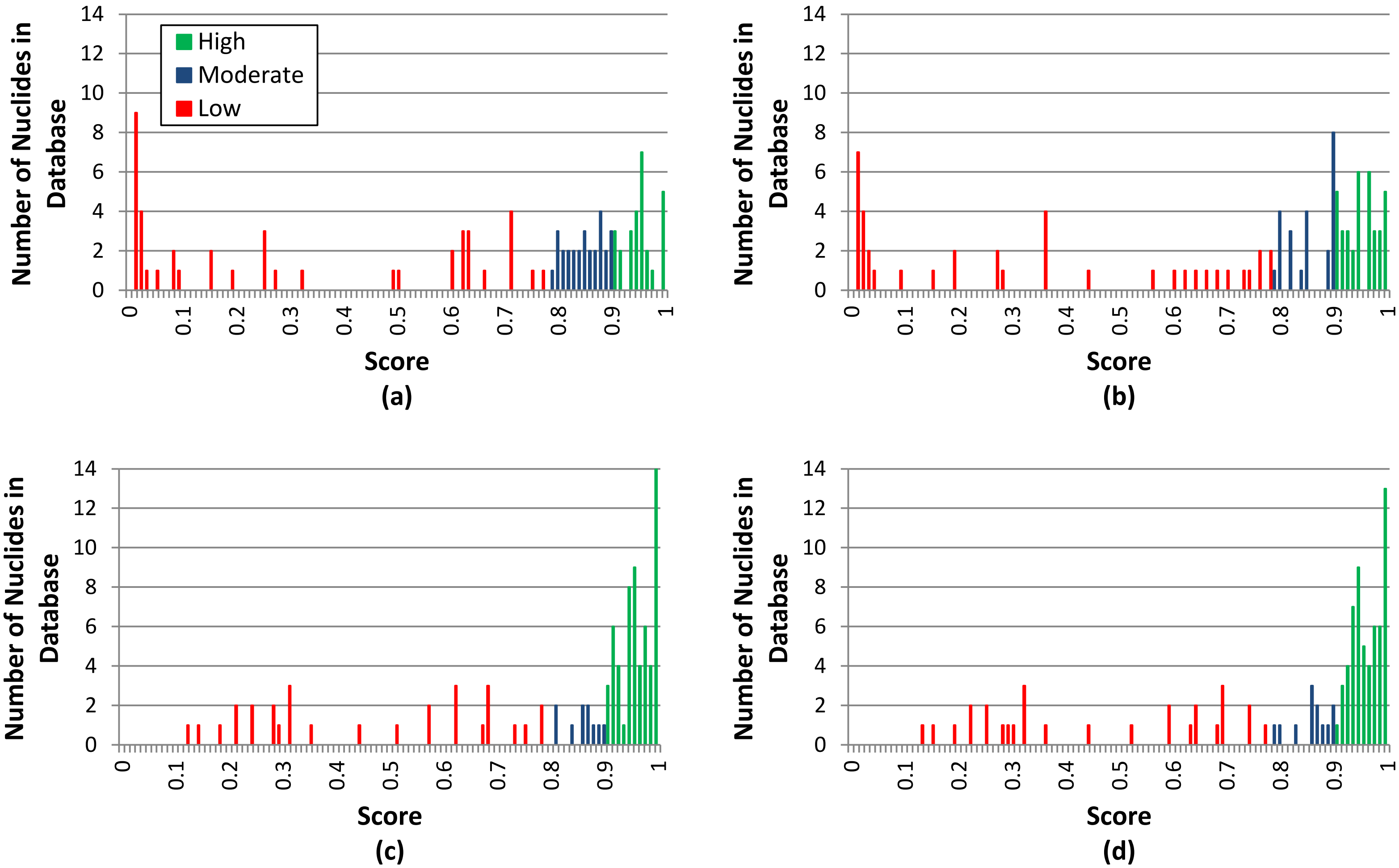


Figure 16. Overall scores for the 98 database nuclides within the (a) Fort Calhoun, (b) Palisades, (c) Saint Lucie Unit 2, and (d) South Texas individual assembly classes.

A number of observations can be made about Figure 7 through Figure 16:

- PWR assembly classes tend to have higher AR than BWR assembly classes.
- Older assembly classes tend to have lower AR than newer assembly classes. An obvious exception to this observation is that the newer GE BWR 4,6 assembly class has a lower AR than the older GE BWR 2,3 assembly class.
- The Fort Calhoun and Palisades individual assembly classes have a lower AR than the CE 16x16 assembly class, even though both reactors use the CE 16x16 reactor design.
- The Saint Lucie Unit 2 assembly class has a AR profile very similar to the CE 14x14 assembly class, and the unit uses the CE 14x14 reactor design.
- The South Texas assembly class has a lower AR than the W 17x17 assembly class, even though both of the South Texas units use the W 17x17 reactor design.
- The W 17x17 assembly class has the highest AR in general.
- The GE BWR 4,6 and the Fort Calhoun assembly classes have the worst low AR profiles, but the Fort Calhoun assembly class has more nuclides that fall into the Moderate and High AR categories.

These observations highlight the usefulness of the methodology for analyzing the database. In separating the reactor models into their assembly classes, the method can highlight details that the initial investigation smeared. The higher AR of the PWR assembly classes makes sense because PWRs use higher enrichments than BWRs and therefore would be much closer to the assumption of 4.0 wt.% enrichment. The higher AR of the newer reactors also makes sense for the same type of reason, while the newer

BWR class has a lower AR because its range of assembly designs grew at both the high and low end.

Another benefit of the method is ability to weigh each individual score based on the historical frequency of that specific assembly design. This affirmed the hypothesis made in subsection 4.2. TABLE VI and TABLE VII show the new overall scores for each nuclide in the database, grouped into the AR classifications. Four nuclides were removed from this table due to the fact that they were not produced in any of the models: ^{244m}Am, ^{241}Np, ^{235}Pa, and ^{241}U. The numerical overall scores of the same can be found in Appendix B.

All of the nuclides included in Figure 5 fall into the high AR, with very few exceptions. Four of these exceptions come from the Fort Calhoun assembly class: ^{234m}Pa, ^{239}Pu, ^{234}Th, and ^{238}U. This seems reasonable given the large difference in both enrichment and initial uranium content between the assembly designs and the database model. The two other exceptions come from the GE BWR 2,3 assembly class: ^{241}Am and ^{240}Pu. As for the hypothesis that more nuclides would move into the higher AR categories, the results shown in the tables greatly emphasize this fact. Beyond the nine nuclides in Figure 5, many more fall into the high and moderate categories, with up to 70 nuclides joining the high category at best.

TABLE VI. Summary of the classifications for each of the database nuclides, by PWR assembly class. (239Am to 251Cm)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Am-239	Moderate	High	High	High	High	High	High
Am-240	High	High	High	High	High	High	High
Am-241	High	High	High	High	High	High	High
Am-242	Moderate	High	High	High	High	High	High
Am-242m	High	High	High	High	High	High	High
Am-243	Low	Low	Moderate	High	Low	Low	High
Am-244	Low	Low	Moderate	High	Low	Low	High
Am-245	Low	Low	Moderate	Moderate	Low	Low	Moderate
Am-246	Low	Low	Low	Moderate	Low	Low	Moderate
Bk-249	Low	Low	Low	Low	Low	Low	Low
Bk-250	Low	Low	Low	Low	Low	Low	Low
Bk-251	Low	Low	Low	Low	Low	Low	Low
Cd-113m	High	High	High	High	High	High	High
Cf-249	Low	Low	Low	Low	Low	Low	Low
Cf-250	Low	Low	Low	Low	Low	Low	Low
Cf-251	Low	Low	Low	Low	Low	Low	Low
Cf-252	Low	Low	Low	Low	Low	Low	Low
Cf-253	Low	Low	Low	Low	Low	Low	Low
Cf-254	Low	Low	Low	Low	Low	Low	Low
Cf-255	Low	Low	Low	Low	Low	Low	Low
Cm-241	Moderate	Moderate	High	High	Moderate	Moderate	High
Cm-242	Moderate	Moderate	High	High	High	High	High
Cm-243	Moderate	Moderate	High	High	Moderate	Moderate	High
Cm-244	Low	Low	Moderate	Moderate	Low	Low	Moderate
Cm-245	Low	Low	Moderate	Moderate	Low	Low	Moderate
Cm-246	Low	Low	Low	Moderate	Low	Low	Moderate
Cm-247	Low	Low	Low	Low	Low	Low	Moderate
Cm-248	Low	Low	Low	Low	Low	Low	Low
Cm-249	Low	Low	Low	Low	Low	Low	Low
Cm-250	Low	Low	Low	Low	Low	Low	Low
Cm-251	Low	Low	Low	Low	Low	Low	Low

TABLE VI. Continued. (^{134}Cs to ^{242}Pu)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Cs-134	High	High	High	High	High	High	High
Cs-135	High	High	High	High	High	High	High
Cs-137	High	High	High	High	High	High	High
Eu-154	High	High	High	High	High	High	High
Eu-155	High	High	High	High	High	High	High
He-4	Moderate	High	High	High	High	High	High
I-129	High	High	High	High	High	High	High
Kr-85	High	High	High	High	High	Moderate	High
Nb-94	Moderate	Moderate	High	High	High	High	High
Np-235	High	High	High	High	High	High	High
Np-236	High	High	High	High	High	High	High
Np-236m	High	High	High	High	High	High	High
Np-237	High	High	High	High	High	High	High
Np-238	High	High	High	High	High	High	High
Np-239	High	High	High	High	High	High	High
Np-240	Moderate	Moderate	High	High	Moderate	Moderate	High
Np-240m	Low	Low	Moderate	Moderate	Low	Low	Moderate
Pa-231	Low	Low	Moderate	Moderate	Low	Low	Moderate
Pa-232	Low	Low	Moderate	Moderate	Moderate	Low	Moderate
Pa-233	High	High	High	High	High	High	High
Pa-234	High	High	High	High	High	High	High
Pa-234m	High	High	High	High	High	High	High
Pd-107	Moderate	Moderate	High	High	Moderate	Moderate	High
Pm-147	High	High	High	High	High	High	High
Pu-236	High	High	High	High	High	High	High
Pu-237	Moderate	High	High	High	Moderate	High	High
Pu-238	High	High	High	High	High	High	High
Pu-239	High	High	High	High	High	High	High
Pu-240	High	High	High	High	High	High	High
Pu-241	High	High	High	High	High	High	High
Pu-242	Moderate	Moderate	High	High	Moderate	Moderate	High

TABLE VI. Continued. (^{243}Pu to ^{93}Zr)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Pu-243	Low	Low	High	High	Low	Moderate	High
Pu-244	Low	Low	Moderate	Moderate	Low	Low	Moderate
Pu-245	Low	Low	Moderate	Moderate	Low	Low	Moderate
Pu-246	Low	Low	Low	Moderate	Low	Low	Moderate
Ru-106	Moderate	Moderate	High	High	High	High	High
Sb-125	High	High	High	High	High	High	High
Se-79	High	High	High	High	High	High	High
Sm-151	High	High	High	High	High	High	High
Sn-126	High	High	High	High	High	High	High
Sr-90	High	High	High	High	High	High	High
Tc-99	High	High	High	High	High	High	High
Th-226	High	High	High	High	Moderate	High	High
Th-227	High	High	High	High	High	High	High
Th-228	High	High	High	High	High	High	High
Th-229	High	High	High	High	Moderate	High	High
Th-230	High	High	High	High	High	High	High
Th-231	Moderate	Moderate	High	High	Moderate	Moderate	High
Th-232	Moderate	Moderate	High	High	Moderate	Moderate	High
Th-233	High	High	High	High	High	High	High
Th-234	High	High	High	High	High	High	High
U-230	High	High	High	High	Moderate	High	High
U-231	High	High	High	High	Moderate	High	High
U-232	High	High	High	High	High	High	High
U-233	Moderate	Moderate	High	High	Moderate	Moderate	High
U-234	High	High	High	High	High	High	High
U-235	Low	Low	Moderate	Moderate	Low	Low	Moderate
U-236	Moderate	Moderate	High	High	Moderate	Moderate	High
U-237	Moderate	High	High	High	High	Moderate	High
U-238	High	High	High	High	High	High	High
U-239	High	High	High	High	High	High	High
U-240	Low	Low	Moderate	Moderate	Low	Low	Moderate
Zr-93	High	High	High	High	High	High	High

TABLE VII. Summary of the classifications for each of the database nuclides, by BWR and individual assembly classes. (239Am to 249Cm)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Am-239	Low	Low	Moderate	Moderate	High	High
Am-240	Low	Low	Moderate	Moderate	High	High
Am-241	Low	High	High	High	High	High
Am-242	Low	Low	Moderate	Moderate	High	High
Am-242m	Low	High	High	High	High	High
Am-243	Low	Low	Low	Low	Low	Moderate
Am-244	Low	Low	Low	Low	Low	Low
Am-245	Moderate	Low	Low	Low	Low	Low
Am-246	Low	Low	Moderate	Low	Low	Low
Bk-249	Low	Low	Low	Low	Low	Low
Bk-250	Low	Low	Low	Low	Low	Low
Bk-251	Low	Low	Low	Low	Low	Low
Cd-113m	Moderate	Moderate	High	High	High	High
Cf-249	Low	Low	Low	Low	Low	Low
Cf-250	Low	Low	Low	Low	Low	Low
Cf-251	Low	Low	Low	Low	Low	Low
Cf-252	Low	Low	Low	Low	Low	Low
Cf-253	Low	Low	Low	Low	Low	Low
Cf-254	Low	Low	Low	Low	Low	Low
Cf-255	Low	Low	Low	Low	Low	Low
Cm-241	Low	Low	Low	Low	Moderate	Moderate
Cm-242	Low	Low	Moderate	Moderate	High	High
Cm-243	Low	Low	Low	Low	Moderate	Moderate
Cm-244	Low	Low	Low	Low	Low	Low
Cm-245	Low	Low	Low	Low	Low	Low
Cm-246	Low	Low	Moderate	Low	Low	Low
Cm-247	Moderate	Low	Low	Low	Low	Low
Cm-248	Low	Low	Low	Low	Low	Low
Cm-249	Low	Low	Low	Low	Low	Low

TABLE VII. Continued. (^{250}Cm to ^{241}Pu)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Cm-250	Low	Low	Low	Low	Low	Low
Cm-251	Low	Low	Low	Low	Low	Low
Cs-134	Moderate	Moderate	Moderate	Moderate	High	High
Cs-135	Moderate	Moderate	Moderate	High	High	High
Cs-137	High	High	High	High	High	High
Eu-154	Moderate	Moderate	Moderate	Moderate	High	High
Eu-155	Moderate	Moderate	Moderate	Moderate	High	High
He-4	Moderate	Moderate	Moderate	Moderate	High	High
I-129	Moderate	High	High	High	High	High
Kr-85	Moderate	Moderate	High	Moderate	High	High
Nb-94	Low	Low	Moderate	Moderate	High	High
Np-235	Moderate	Low	Low	High	High	High
Np-236	High	High	High	High	High	High
Np-236m	High	Moderate	Moderate	High	High	High
Np-237	High	High	High	High	High	High
Np-238	High	Moderate	Moderate	High	High	High
Np-239	Moderate	Moderate	Moderate	Moderate	High	High
Np-240	Low	Low	Low	Low	Moderate	Moderate
Np-240m	Low	Low	Low	Low	Low	Low
Pa-231	Low	Low	Low	Low	Low	Low
Pa-232	Low	Low	Low	Low	Moderate	Moderate
Pa-233	High	High	High	High	High	High
Pa-234	High	Moderate	Moderate	High	High	High
Pa-234m	High	High	Moderate	High	High	High
Pd-107	Low	Low	Moderate	Moderate	Moderate	Moderate
Pm-147	High	Moderate	Moderate	Moderate	High	High
Pu-236	Moderate	Moderate	Moderate	High	High	High
Pu-237	Low	Low	Low	Low	High	High
Pu-238	Low	Moderate	Moderate	Moderate	High	High
Pu-239	High	High	Moderate	High	High	High
Pu-240	Moderate	High	High	High	High	High
Pu-241	Low	Moderate	High	High	High	High

TABLE VII. Continued. (^{242}Pu to ^{93}Zr)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Pu-242	Low	Low	Low	Low	Moderate	Moderate
Pu-243	Low	Low	Low	Low	Moderate	Moderate
Pu-244	Low	Low	Low	Low	Low	Low
Pu-245	Moderate	Low	Low	Low	Low	Low
Pu-246	Low	Low	Low	Low	Low	Low
Ru-106	Low	Low	Moderate	Moderate	High	High
Sb-125	Moderate	Moderate	High	High	High	High
Se-79	High	High	High	High	High	High
Sm-151	High	High	High	High	High	High
Sn-126	Moderate	Moderate	High	High	High	High
Sr-90	Moderate	Moderate	High	Moderate	High	High
Tc-99	High	High	High	High	High	High
Th-226	Moderate	Low	Low	Moderate	High	High
Th-227	Low	Moderate	High	High	High	High
Th-228	High	High	Moderate	High	High	High
Th-229	Moderate	Low	Low	Moderate	High	High
Th-230	High	High	High	High	High	High
Th-231	Low	Low	Low	Low	Moderate	Moderate
Th-232	Moderate	Moderate	Moderate	Moderate	High	Moderate
Th-233	High	High	High	High	High	High
Th-234	High	High	Moderate	High	High	High
U-230	Moderate	Low	Low	Moderate	High	High
U-231	Moderate	Low	Low	Moderate	High	High
U-232	High	High	Moderate	High	High	High
U-233	Low	Low	Low	Low	Moderate	Moderate
U-234	High	High	High	High	High	High
U-235	Low	Low	Low	Low	Low	Low
U-236	Moderate	Low	Moderate	Low	Moderate	Moderate
U-237	High	High	High	Moderate	High	High
U-238	High	High	Moderate	High	High	High
U-239	Moderate	Moderate	Moderate	Moderate	High	High
U-240	Low	Low	Low	Low	Low	Low
Zr-93	High	High	High	High	High	High

The applicability of the SFD to certain systems and certain studies is clearly laid out in TABLE VI and TABLE VII. The database is a much better estimator of newer, PWR spent fuel compositions than for either older or BWR systems. The database is able to estimate masses for the current fission products under consideration very well, but it is not able to estimate masses for some higher actinides in any sort of reliable manner. From this data, some broad observations can be made about what the database is not able to predict well.

Berkelium falls into the low classification in every category. That the database does not estimate these nuclides well is not of much concern because they have very low thermal cross sections and relatively short half-lives.[24] Therefore, they will quickly decay away in waste storage, much faster than would be possible to recycle the fuel and extract the berkelium nuclides for research. Moreover, the low AR for Berkelium or any other radionuclide should not affect the burnup during reactor operation, given the high AR for ^{137}Cs, which increases linearly with burnup.

Likewise, californium falls into the low classification across the board. This could affect the overall accuracy of storage or reprocessing evaluations made using the SFD since certain californium nuclides emit neutrons and have high thermal cross sections.[24] Furthermore, ^{249}Cf and ^{251}Cf have half-lives in the hundreds of years, so they would be present during reprocessing. However, because californium is produced in such small quantities, from 10^{-7} to 10^{-5} grams per assembly, these effects should be negligible.

Curium falls into the low category for almost all of the BWR and individual assembly classes, and for the PWR assembly classes, as the mass number increases, curium moves from the high AR to the low AR classification. Some curium nuclides are long-lived, and some have substantial thermal cross sections.[24] This could also affect reprocessing studies.

Almost all of the americium nuclides fall into the low category for the BWR assembly classes. Some americium nuclides are long-lived, and some have substantial thermal cross sections.[24] The GE BWR 4,6 assembly class does have two important nuclides that fall into the high classification, ^{241}Am and ^{242m}Am, but the older 2,3 assembly class only has a quickly decaying nuclide that falls above the low category.

Finally, the BWR assembly classes have some plutonium nuclides that lie in the high category and others that lie in the low category. The most important, ^{239}Pu, falls into the high category for both classes. However, most of the other plutonium nuclides have high thermal cross sections and/or long half-lives.[24] This would affect waste management studies since plutonium is the most commonly separated element during reprocessing.

Future work should delve into the topic of understanding how variations in curium might affect the overall accuracy of the database. Furthermore, future work should look into how the low AR of americium and the variable AR of plutonium could affect general BWR waste models created using the database. With caution informed by these caveats and the specific categorizations laid out in TABLE VI and TABLE VII, the SFD can be a useful tool for future studies.

5. ASSESSMENT OF USED FUEL INVENTORIES

The SFD was designed to be used in broader advanced fuel cycle and waste management studies, and an initial assessment of the inventories reiterates the need for these studies in a time of changing policy.

After Yucca Mountain was abandoned as the chosen solution to deal with the growing pile of used fuel, the Obama Administration called for a Blue Ribbon Commission to provide advice and to make recommendations for a new waste management strategy.[3] Based on their report, the Department of Energy developed a strategy that would set up a pilot interim storage facility, then a larger, full-scale interim storage facility, and finally a new long-term repository built with the support of the local population.[25] The Department of Energy would give priority to accepting used fuel from shutdown reactors at the pilot interim storage facility; so if there were room left over, the question would be which fuel is the best choice for second priority?

Moreover, the Administration accepts the caveat given by the Blue Ribbon Commission to maintain flexibility in anticipation of advanced fuel cycles.[25] However, a team from Oak Ridge National Laboratory suggested that if future cycles incorporated reprocessing, the waste produced in the future would be more than enough to satisfy fast reactor fuel needs and that all of the current spent fuel could be placed in a repository, reserving only small portions for research.[20]

Section 5 seeks to consider what lessons can be drawn from the database to make further recommendations. The first two subsections focus on a numerical understanding of the inventories in the database. Since waste management is the main concern, the data is considered in terms of radiotoxicity and of heat load. The radiotoxicity is important because it determines the amount of sheilding needed to protect workers and the public, and the heat produced by the used fuel is important because it determines the space and cooling requirements for any storage facility. The third subsection uses the information provided in the first two to make suggestions regarding the storage, transmutation, and research of used nuclear fuel.

5.1. Radiotoxicity

Radiotoxicity is a measure of the potential harm to a human body from a radioactive substance. It is directly related to the radioactivity of the substance and can be thought of as "the volume of water needed to dilute a radionuclide so that it would be safe to drink."[26] The radiotoxicity from a certain nuclide is defined in Eq. (5).[27]

$$R_i(t) = F_{d,i} \times \lambda_i N_i(t) \tag{5}$$

where $R_i(t)$ is the radiotoxicity of nuclide i at time t, $F_{d,i}$ is the factor of dose for nuclide i, λ_i is the nuclide's decay constant, and $N_i(t)$ is the number of atoms of nuclide i present in the substance at time t. The factors of dose for various nuclides can be found in publications by the International Commission on Radiological Protection (ICRP), which list factors of dose for both inhalation and ingestion of the nuclide.[28] A simple summation over all of the nuclides in the substance will find its total radiotoxicity.

To calculate the radiotoxicity of the estimates found in the Spent Fuel Database, a MATLAB script was written to quickly move through the data. The mass estimates were imported from the Excel file holding the database and were converted to the number of atoms using Eq. (6).

$$N_i(t) = \frac{m_i(t)A_v}{M_i} \tag{6}$$

where $m_i(t)$ is the mass of nuclide i in grams, A_v is Avogadro's number, and M_i is the molar mass of nuclide i in grams per mole. The radiotoxicity was calculated both for the total radiotoxicity of all the spent fuel from operating reactors in one year and for the average radiotoxicity from the different types of assembly designs.

Figure 17 presents the radiotoxicity of all the spent fuel from operating reactors in one year for both ingestion and inhalation at each of the reference times used in the SFD. For this calculation, the assumption was made that each of the reactors in the database removed one-third of the reactor core in that year, rounding up to the nearest whole number of assemblies removed. Also, whenever the ICRP listed more than one factor for inhalation, based on a slow or fast intake of the nuclide, the higher factor of dose was chosen to be conservative. Figure 17 shows that the radiotoxicity of all the spent fuel together decreases by almost a factor of ten for ingestion over the time periods included in the database. However, for inhalation, the radiotoxicity only decreases by about 50%.

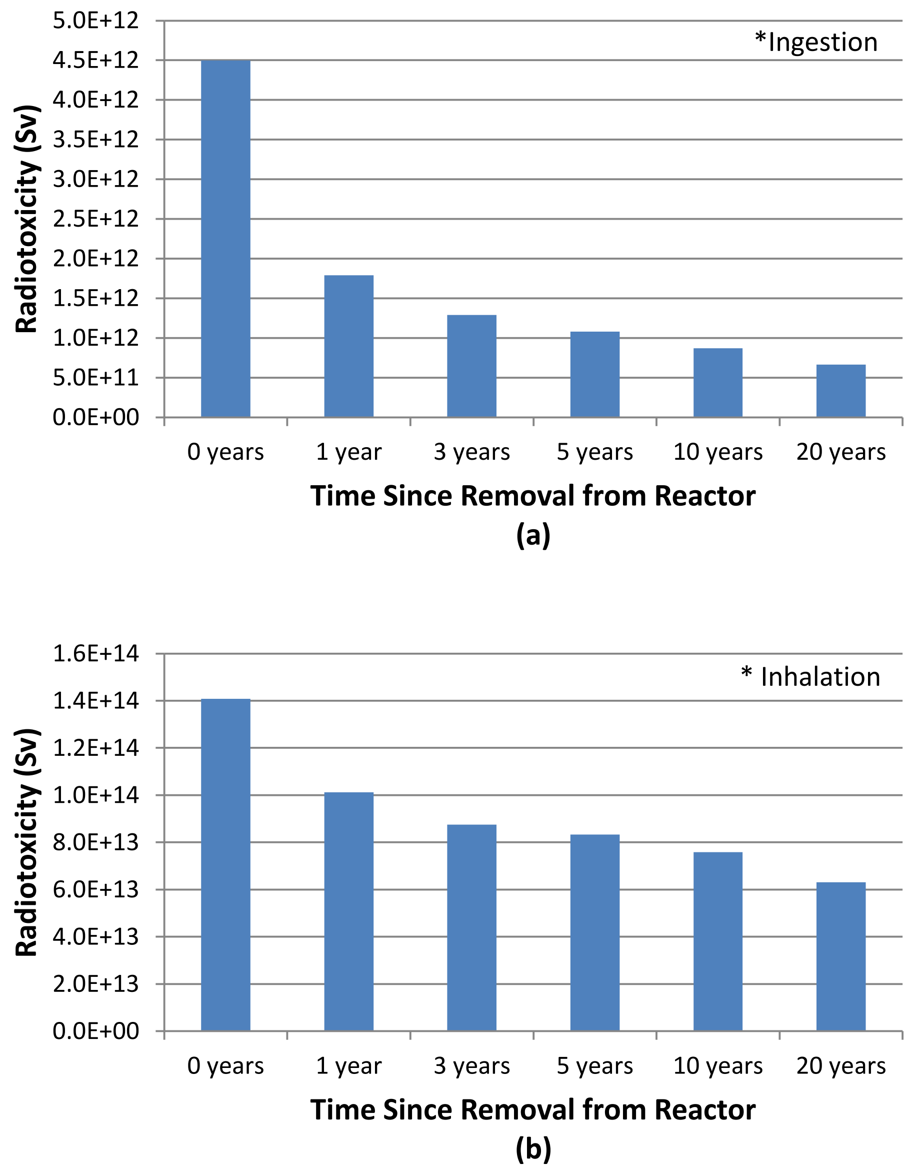


Figure 17. Total radiotoxicity of all spent fuel from operating reactors for (a) ingestion and (b) inhalation for each of the reference times used in the Spent Fuel Database.

Figure 18 shows the composition of the "20 years" radiotoxicity bars in Figure 17. For radiotoxicity by ingestion, two fission products, ^{90}Sr and ^{137}Cs, contribute the most, comprising about 50% of the risk. For both ingestion and inhalation, five actinides also contribute greatly: ^{238}Pu, ^{241}Am, ^{244}Cm, ^{241}Pu, and ^{240}Pu. For inhalation, these five present approximately 97% of the radiotoxic risk.

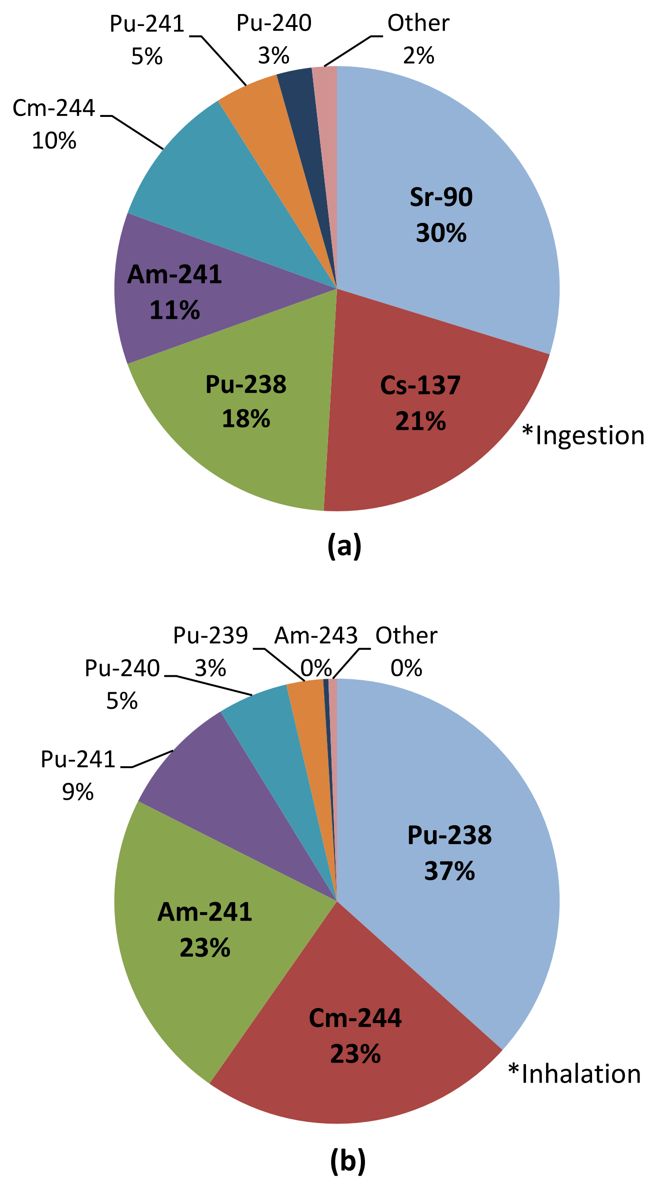
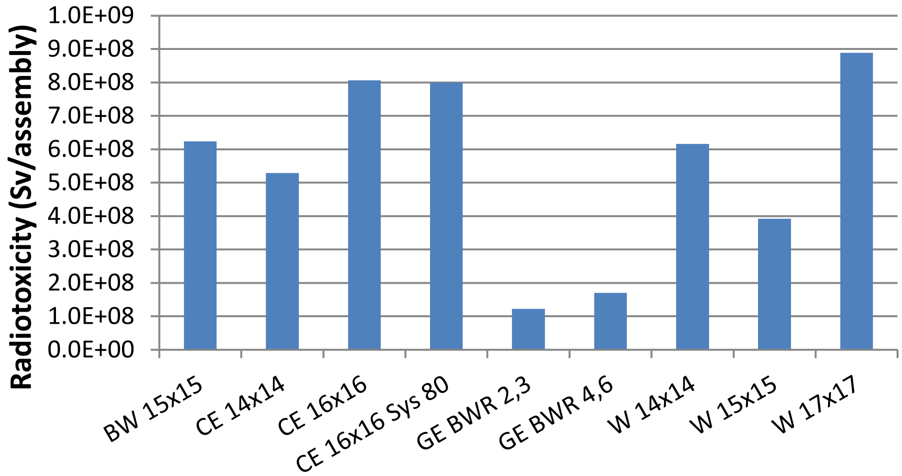
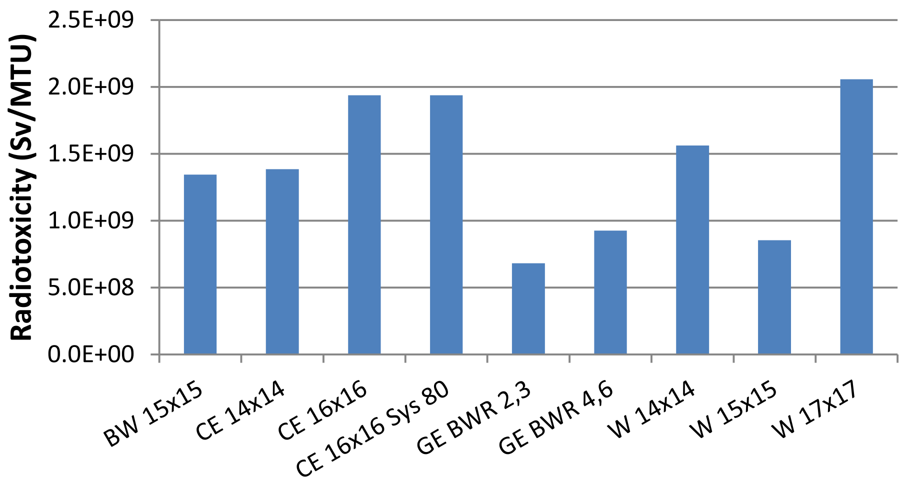


Figure 18. Composition of the total radiotoxicity of all spent fuel from operating reactors for (a) ingestion and (b) inhalation after 20 years.



(a)



(b)

Figure 19. Average radiotoxicity of an assembly upon removal from the core, by assembly class, in (a) Sv/assembly and (b) Sv/MTU.

Figure 19 presents the average radiotoxicity for each assembly class upon removal from the reactor. The chart is shown using units of Sievert per assembly (Sv/assembly) and Sievert per metric tonne of uranium (Sv/MTU). The latter is the traditional unit used to compare radiotoxicity across various media and is useful when considering reprocessing and transmutation. However, the former unit was also used because when the spent fuel is put into storage, the assembly will remain intact, so this unit is useful for comparison as well.

The charts in Figure 19 show similar trends. The highest radiotoxicity would come from a W 17x17 assembly, and the BWR assemblies would have lower toxicities. The newer PWR assemblies generally have higher toxicities than older assemblies. These trends make sense given the difference in enrichment levels. However, because BWR assemblies are much smaller than PWR assemblies, on a per assembly basis, the risk is much lower. Furthermore, a B&W 15x15 assembly would have a higher radiotoxicity than a CE 14x14 assembly, but this relationship is reversed when considering it on a per MTU basis. This sort of difference is relevant to how the used fuel is to be managed in the future.

5.2. Heat Load

The heat emanating from the used fuel comes from the decay of fission products and transuranic isotopes. Formally, the heat would be calculated using standard knowledge of branching ratios and energy release, along with the calculated activities of all the isotopes in a given substance. This calculation is summarized in Eq. (7).[29]

$$H(t) = \sum_{i=1}^{M} \left(E_{\alpha}^{i} + E_{\beta}^{i} + E_{\gamma}^{i}\right)\lambda_i N_i(t) \qquad (7)$$

where $H(t)$ is the total decay heat as a function of time, M is the number of radioactive isotopes in the substance, and E_{α}^{i}, E_{β}^{i}, and E_{γ}^{i} are the mean alpha, beta, and gamma energy releases per disintegration. Fortunately, ORIGEN-ARP has this capability built in, so to calculate the heat coming from the spent fuel in the database required only one change in each input deck: replacing kilograms with watts as the unit of interest for the output. The heat load was calculated both for the total heat from all of the spent fuel from operating reactors in one year and for the average heat load from the generic types based on assembly class.

Figure 20 presents the heat load from all of the spent fuel in one year, for each of the reference years used in the database. This calculation assumed that one-third of each reactor core was removed in the specified year, as was assumed for the radiotoxicity analysis. The vertical axis is shown using a logarithmic scale to ensure that the magnitude of the newly removed used fuel did not swamp the smaller numbers in the graph. The total heat load is also written above each bar for clarity. Figure 20 shows that the total heat load dramatically decreases over the time range considered by the database. After 20 years, the heat load from the spent fuel from all of the reactors is almost on par with that of one operating reactor.

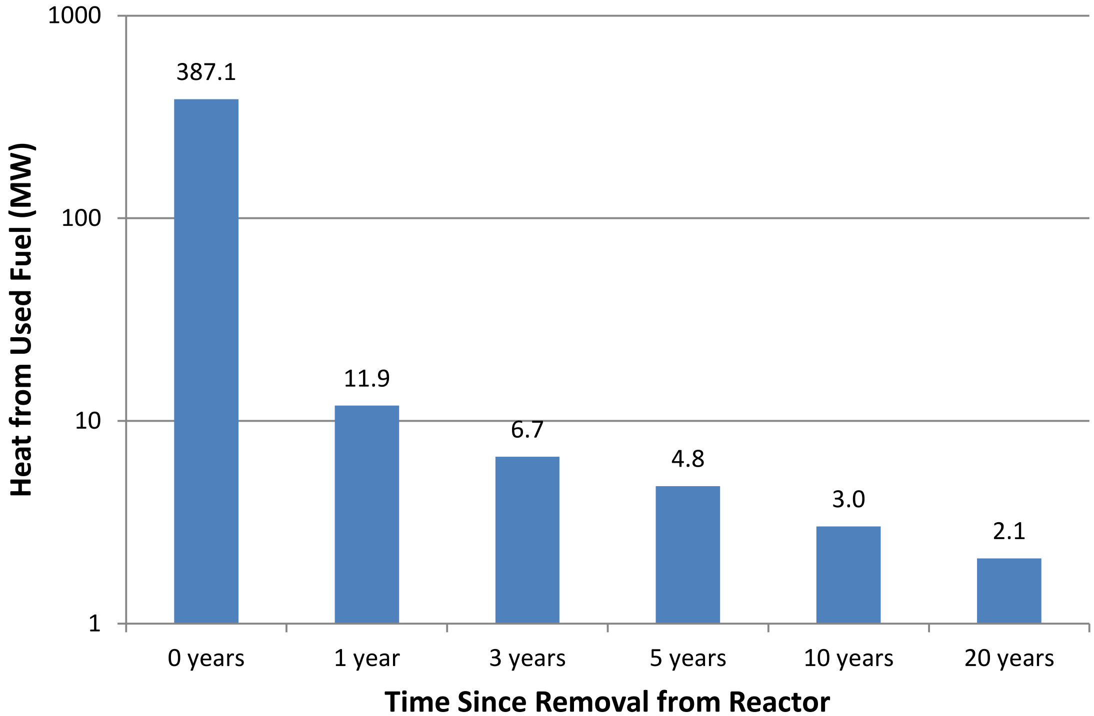
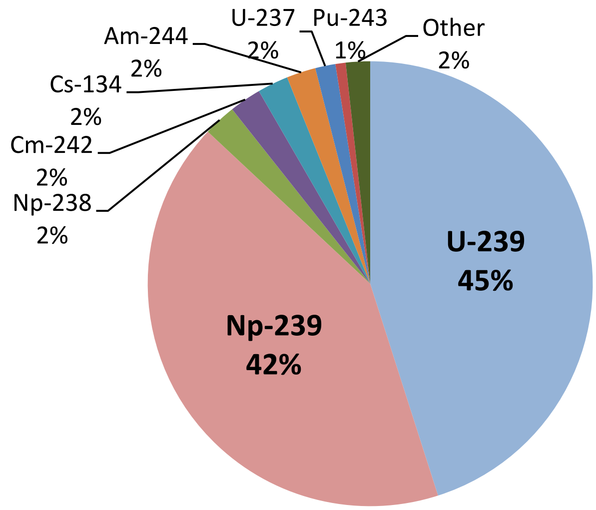
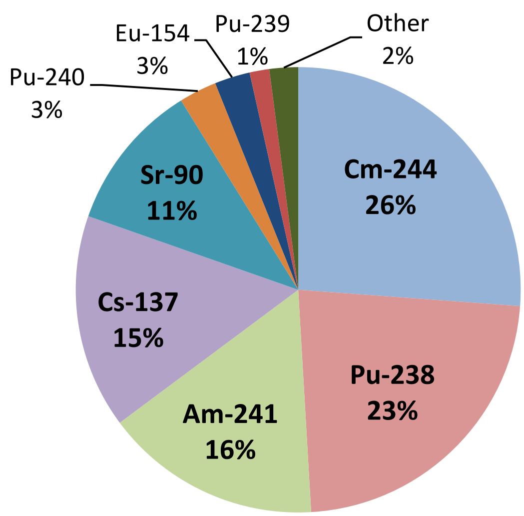


Figure 20. Total heat load of all the spent fuel from operating reactors for each of the reference times used in the Spent Fuel Database, shown with a logarithmic vertical axis.

Figure 21 shows the composition of the sources producing the heat load presented in Figure 20. The first pie-chart depicts the composition from 0 years, and the second depicts the composition from 20 years. For the newly removed fuel, ^{239}Np and ^{239}U produce the majority of the 387.1 MW. After 20 years, ^{241}Am, ^{244}Cm, ^{137}Cs, ^{238}Pu, and ^{90}Sr contribute the most to the heat load.



(a)



(b)

Figure 21. Composition of the total heat load of all the spent fuel from operating reactors after (a) 0 years and after (b) 20 years.

Figure 22 presents the average heat load of a generic assembly recently taken out of the reactor, categorized by assembly class. The shape of the bar chart strongly resembles Figure 19(a) since it uses the same basis: one assembly. Once again, the W

17x17 assembly will produce the most heat, and the BWR assemblies will produce the least. The newer designs produce more heat than the older designs. The same explanations used to understand Figure 19 are applicable here, so the relationships between the assembly classes can be used for either radiotoxicity or heat load. This is expected since both are related to the radioactivity of the fuel. Moreover, the relationships shown in Figure 22 do not noticeably change over the 20 years considered by the database.

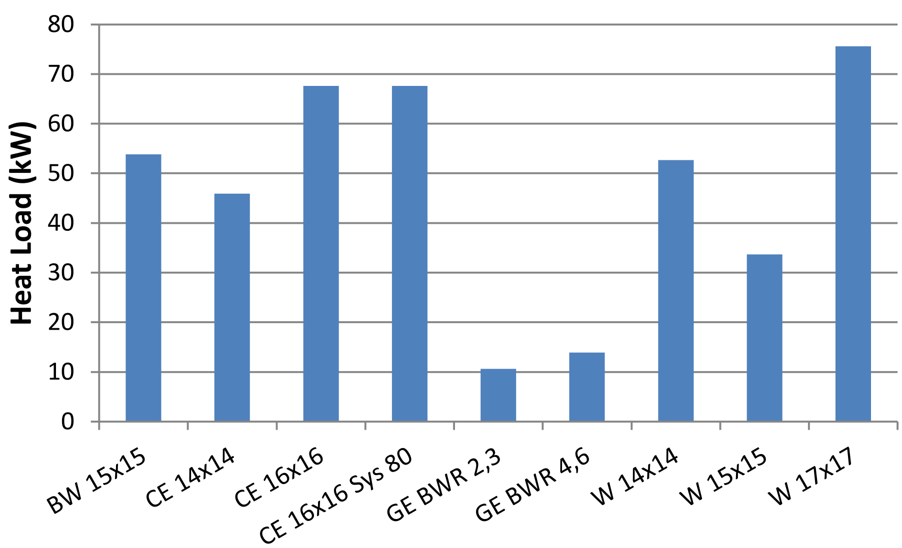


Figure 22. Average assembly heat load upon removal from the core, by assembly class.

5.3. Suggestions for Storage, Transmutation, and Research Activities

Based on the information provided in the previous two subsections, some suggestions can be made for policy makers and as a starting point for future research. For interim and long-term storage, using the assembly as the basis is the most suitable since the used fuel generally would not be removed from it. With this in mind, a few suggestions are put forth:

- If the pilot interim storage facility has any vacancies after receiving all of the used fuel from shutdown reactors, it should next prioritize accepting the oldest used fuel from a GE BWR 2,3 type of reactor. This would minimize the heat load and the potential radiotoxic risk.
- With an expected capacity of only 20,000 MTHM,[25] the full-scale interim storage facility should also prioritize accepting the oldest used fuel from operating reactors. Priority should be given to fuel from the two BWR assembly classes, to the W 15x15 class, and then to the CE 14x14 class.
- If reprocessing is incorporated into the U.S. fuel cycle, simply separating the plutonium from the fuel would decrease the heat load from the oldest assemblies by about 25%. This could save a lot of space.

Finally, one last suggestion is put forth in relation to the Oak Ridge team's recommendation to set aside some of the used fuel for research purposes. Because the composition of the fuel and the sources of heat and radiotoxicity change so much even over twenty years, the fuel that is set aside should reflect the larger dynamics, from the oldest fuel to the hottest, most toxic used fuel.

6. CONCLUSIONS

The goal of this research was to develop a methodology to collect inventory estimates for the analysis and characterization of used fuel in the United States. Once collected the inventory estimates were stored in a Microsoft Excel file, creating the Spent Fuel Database, a tool for advanced fuel cycle and waste management studies. The research was broken into four parts: data collection, model development, analysis, and assessment.

The data that was collected was largely drawn from publicly available information provided by the Nuclear Regulatory Commission and the Energy Information Administration. The collection gathered data on the 103 operating reactors in January 2012 and focused on reactor type and design, fuel type, month of reactor startup, initial power, subsequent power uprates, number of fuel assemblies in the core, and capacity factors. However, three key factors could not be obtained: historic enrichment values, historic initial uranium contents, and capacity factors before 2003 for each individual reactor.

Using the collected data, individual plant models were created using the graphical user interface of ORIGEN-ARP. The models were built with a set of assumptions intended to fill in the information gaps. The fuel enrichment was assumed to be 4.0 wt.% for all of the reactors. The initial uranium content was chosen by assuming the most likely value for each unit. The available capacity factors for each individual reactor were assumed to be more representative of its history than industry averages. It was also assumed that the reactor operated at 100% power whenever it was not shut down for refueling and that the shutdowns occurred every 1.5 years. The output for each reactor included current inventory estimates for used fuel taken out of the reactor 0, 1, 3, 5, 10, and 20 years ago.

The analysis of the inventory estimates centered on an idea of an applicability range that was defined as the degree to which a correct estimate can be made quantitatively. For the database, it was used as a method to compare the database estimates with mass estimates produced using knowledge of past fuel assembly designs for general reactor classes. The differences were categorized into three classifications: high AR, moderate AR, and low AR. In terms of reactor types, PWRs were shown to have a much higher AR than BWRs, and older assembly classes were shown to have a lower AR than newer classes. In terms of specific nuclides, the fission products in the database were shown to consistently have a high AR. However, berkelium and californium had low AR for all of the assembly classes, curium had low AR for BWR classes and mixed AR for PWR classes, and americium and some plutonium isotopes had low AR for BWR classes.

Finally, the assessment of the inventory estimates considered the potential radiotoxicity and heat load from these masses. While the radiotoxicity by ingestion decreased by about a factor of 10, the radiotoxicity by inhalation only decreased by about 50% from newest used fuel to the oldest. Conversely, the heat load decreased significantly over the same time frame. On a per assembly basis, the radiotoxicity and heat load showed similar trends, with newer PWR assemblies being the highest and

BWR assemblies being the lowest in both categories. With these results in mind, a number of suggestions were made relating to current and possible waste management strategies. At a potential interim storage facility, priority should be given to the oldest BWR assemblies to reduce the radiotoxic risk and heating requirements, followed by Westinghouse 15x15 assemblies and then Combustion Engineering 14x14 assemblies. Reprocessing and transmuting as much of the used fuel as possible is highly encouraged to reduce the radiotoxicity and heat of the waste entering storage.

This research accomplished its goal of creating an easily accessible database for advanced fuel cycle and waste management studies. To continue improving the SFD, future work should seek to address the challenges described in the preceding sections. First, the magnitude of the impact of the variations in AR for curium for all assembly classes and the low AR of americium and the mixed AR of plutonium for BWR classes should be quantified. Next, the reactor models should be rerun using more historically representative enrichment values to improve the overall AR of the database. Finally, another large effort is needed to incorporate the used fuel of all the shutdown reactors into the database. Even in its current form, though, the Spent Fuel Database is a useful tool for reference.

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APPENDIX A

TABLE VIII gives the complete list of reactor initial powers and uprates.[15, 17] TABLE IX gives the complete list of assemblies for each reactor.[17] TABLE X lists capacity factors for each reactor from 2003 to 2010.[18]

TABLE VIII. Reactor uprate dates and power levels.

Name	Reactor Start	1st Thermal Power (MWt)	Month of Uprate	2nd Thermal Power (MWt)	Month of Uprate	3rd Thermal Power (MWt)	Month of Uprate	Current Thermal Power (MWt)
CE14x14: 5 units	**Oct-77**	**2560**					**Sep-88**	**2715**
Calvert Cliffs 1	May-75	2560	Sep-77	2700	--	--	Jul-09	2737
Calvert Cliffs 2	Apr-77	2560	Oct-77	2700	--	--	Jul-09	2737
Millstone 2	Dec-75	2560	--	--	--	--	Jun-79	2700
Saint Lucie 1	Dec-76	2560	--	--	--	--	Nov-81	2700
Saint Lucie 2	Aug-83	2560	--	--	--	--	Mar-85	2700
CE16x16: 9 units	**Apr-82**	**3148**					**Mar-00**	**3295**
Arkansas Nuclear 2	Mar-80	2815	--	--	--	--	Apr-02	3026
Fort Calhoun	Sep-73	1420	--	--	--	--	Aug-80	1500
Palisades	Dec-71	2530	--	--	--	--	Jun-04	2565
Palo Verde 1	Jan-86	3800	May-96	3876	--	--	Nov-05	3990
Palo Verde 2	Sep-86	3800	May-96	3876	--	--	Sep-03	3990
Palo Verde 3	Jan-88	3800	May-96	3876	--	--	Nov-05	3990
San Onofre 2	Aug-83	3390	--	--	--	--	Jul-01	3438
San Onofre 3	Apr-84	3390	--	--	--	--	Jul-01	3438
Waterford 3	Sep-85	3390	Mar-02	3441	--	--	Apr-05	3716
GE7x7-0: 8 units	**Aug-71**	**2191**					**Aug-01**	**2426**
Dresden 2	Jun-70	2527	--	--	--	--	Dec-01	2957
Dresden 3	Nov-71	2527	--	--	--	--	Dec-01	2957
Monticello	Jun-71	1670	--	--	--	--	Sep-98	1775

Nine Mile Point 1	Dec-69	1850	--	--	--	--		1850
Oyster Creek	Dec-69	1930	--	--	--	--		1930
Pilgrim 1	Dec-72	1998	--	--	--	--	May-03	2028
Quad Cities 1	Feb-73	2511	--	--	--	--	Dec-01	2957
Quad Cities 2	Mar-73	2511	--	--	--	--	Dec-01	2957
GE8x8-4: 27 units	**Mar-81**	**2980**	**Jun-92**	**3146**			**Dec-03**	**3263**
Browns Ferry 1	Aug-74	3293	--	--	--	--	Mar-07	3458
Browns Ferry 2	Mar-75	3294	--	--	--	--	Sep-98	3458
Browns Ferry 3	Mar-77	3294	--	--	--	--	Sep-98	3458
Brunswick 1	Mar-77	2436	Oct-96	2558	--	--	May-02	2923
Brunswick 2	Nov-75	2436	Nov-96	2558	--	--	May-02	2923
Clinton	Nov-87	2894	Apr-02	--	--	--		3473
Columbia Generating Station	Dec-84	3323	May-95	--	--	--		3486
Cooper	Jul-74	2381	--	--	--	--	Jun-08	2419
Duane Arnold	Feb-75	1599	Mar-85	1664	--	--	Nov-01	1912
Fermi 2	Jan-88	3293	--	--	--	--	Sep-92	3430
FitzPatrick	Jul-75	2436	--	--	--	--	Dec-96	2536
Grand Gulf 1	Jul-85	3833	--	--	--	--	Oct-02	3898
Hatch 1	Dec-75	2436	Aug-95	2558	Oct-98	2763	Sep-03	2804
Hatch 2	Sep-79	2436	Aug-95	2558	Oct-98	2763	Sep-03	2804
Hope Creek 1	Dec-86	3293	Jul-01	3339	--	--	May-08	3840
La Salle 1	Jan-84	3323	May-00	3489	--	--	Sep-10	3546
La Salle 2	Oct-84	3323	May-00	3489	--	--	Sep-10	3546
Limerick 1	Feb-86	3293	Jan-96	3458	--	--	Apr-11	3515
Limerick 2	Jan-90	3293	Feb-95	3458	--	--	Apr-11	3515
Nine Mile Point 2	Mar-88	3323	--	--	--	--	Apr-95	3467
Peach Bottom 2	Jul-74	3293	Oct-94	3458	--	--	Nov-02	3514
Peach Bottom 3	Dec-74	3293	Jul-95	3458	--	--	Nov-02	3514
Perry 1	Nov-87	3580	--	--	--	--	Jun-00	3758
River Bend 1	Jun-86	2894	Jan-00	3039	--	--	Jan-03	3091
Susquehanna 1	Jun-83	3293	Feb-95	3441	Jul-01	3489	Jan-08	3952
Susquehanna 2	Feb-85	3293	Apr-94	3441	Jul-01	3489	Jan-08	3952
Vermont Yankee	Nov-72	1593	--	--	--	--	Mar-06	1912
W14x14: 6 units	**Dec-72**	**1488**					**Feb-07**	**1664**
Ginna	Jul-70	1520	--	--	--	--	Jul-06	1775

Kewaunee	Jun-74	1650	Jul-03	1673	--	--	Feb-04	1772
Point Beach 1	Dec-70	1259	Nov-02	1280	--	--	May-11	1540
Point Beach 2	Oct-72	1259	Nov-02	1280	--	--	May-11	1540
Prairie Island 1	Dec-73	1620	--	--	--	--	Aug-10	1677
Prairie Island 2	Dec-74	1620	--	--	--	--	Aug-10	1677
W15x15: 15 units	**Aug-74**	**2627**					**Nov-99**	**2710**
Arkansas Nuclear 1	Dec-74	2568	--	--	--	--		2568
Crystal River 3	Mar-77	2452	Jul-81	2544	Dec-02	2568	Dec-07	2609
D.C. Cook 1	Aug-75	3250	--	--	--	--	Dec-02	3304
Davis-Besse	Jul-78	2772	--	--	--	--	Jun-08	2817
Indian Point 2	Aug-74	3071	May-03	3114	--	--	Oct-04	3216
Indian Point 3	Aug-76	3025	Nov-02	3067	--	--	Mar-05	3216
Oconee 1	Jul-73	2568	--	--	--	--		2568
Oconee 2	Sep-74	2568	--	--	--	--		2568
Oconee 3	Dec-74	2568	--	--	--	--		2568
Robinson 2	Mar-71	2200	Jun-79	2300	--	--	Nov-02	2339
Surry 1	Dec-72	2711	Aug-95	2816	--	--	Sep-10	2857
Surry 2	May-73	2711	Aug-95	2816	--	--	Sep-10	2857
Three Mile Island 1	Sep-74	2535	--	--	--	--	Jul-88	2568
Turkey Point 3	Dec-72	2200	--	--	--	--	Sep-96	2300
Turkey Point 4	Sep-73	2200	--	--	--	--	Sep-96	2300
W17x17: 34 units	**Apr-85**	**3260**					**Dec-00**	**3384**
Beaver Valley 1	Oct-76	2652	Sep-01	2689	--	--	Jul-06	2900
Beaver Valley 2	Nov-87	2652	Sep-01	2689	--	--	Jul-06	2900
Braidwood 1	Jul-88	3417	--	--	--	--	May-01	3586
Braidwood 2	Oct-88	3417	--	--	--	--	May-01	3586
Byron 1	Sep-85	3417	--	--	--	--	May-01	3586
Byron 2	Aug-87	3417	--	--	--	--	May-01	3586
Callaway	Dec-84	3411	--	--	--	--	Mar-88	3565
Catawba 1	Jun-85	3411	--	--	--	--		3411
Catawba 2	Aug-86	3411	--	--	--	--		3411
Comanche Peak 1	Aug-90	3411	Oct-01	3458	--	--	Jun-08	3612
Comanche Peak 2	Aug-93	3257	Sep-99	3291	Oct-01	3304	Jun-08	3458
D.C. Cook 2	Jul-78	3411	--	--	--	--	May-03	3468
Diablo Canyon 1	May-85	3338	--	--	--	--	Oct-00	3411

Diablo Canyon 2	Mar-86	3411	--	--	--	--		3411
Farley 1	Dec-77	2637	--	--	--	--	Apr-98	2775
Farley 2	Jul-81	2637	--	--	--	--	Apr-98	2775
Harris 1	May-87	2762	--	--	--	--	Oct-01	2900
McGuire 1	Dec-81	3411	--	--	--	--		3411
McGuire 2	Mar-84	3411	--	--	--	--		3411
Millstone 3	Apr-86	3411	--	--	--	--	Aug-08	3650
North Anna 1	Jun-78	2775	Aug-86	2893	--	--	Oct-09	2940
North Anna 2	Dec-80	2775	Aug-86	2893	--	--	Oct-09	2940
Salem 1	Jun-77	3338	Feb-86	3411	--	--	May-01	3459
Salem 2	Oct-81	3411	--	--	--	--	May-01	3459
Seabrook 1	Aug-90	3411	Feb-05	3587	--	--	May-06	3648
Sequoyah 1	Jul-81	3411	--	--	--	--	Apr-02	3455
Sequoyah 2	Jun-82	3411	--	--	--	--	Apr-02	3455
South Texas 1	Aug-88	3800	--	--	--	--	Apr-02	3853
South Texas 2	Jun-89	3800	--	--	--	--	Apr-02	3853
Summer	Jan-84	2775	--	--	--	--	Apr-96	2900
Vogtle 1	Jun-87	3411	Mar-93	3565	--	--	Feb-08	3625
Vogtle 2	May-89	3411	Mar-93	3565	--	--	Feb-08	3625
Watts Bar 1	May-96	3411	--	--	--	--	Jan-01	3459
Wolf Creek 1	Sep-85	3411	--	--	--	--	Nov-93	3565

TABLE IX. Assemblies per reactor with the mass of initial uranium included.

Name	Number of assemblies in core	Mass of initial Uranium per assembly (kg)
CE14x14: 5 units avg.	**217**	**382**
Calvert Cliffs 1	217	382
Calvert Cliffs 2	217	382
Millstone 2	217	382
Saint Lucie 1	217	382
Saint Lucie 2	217	382
CE16x16: 9 units avg.	**241**	**415**
Arkansas Nuclear 2	177	416
Fort Calhoun	133	416

Palisades	204	416
Palo Verde 1	241	413
Palo Verde 2	241	413
Palo Verde 3	241	413
San Onofre 2	217	416
San Onofre 3	217	416
Waterford 3	217	416
GE7x7-0: 8 units avg.	**724**	**180**
Dresden 2	724	111
Dresden 3	724	188
Monticello	484	188
Nine Mile Point 1	532	193
Oyster Creek	560	193
Pilgrim 1	580	188
Quad Cities 1	724	188
Quad Cities 2	724	188
GE8x8-4: 27 units avg.	**764**	**184**
Browns Ferry 1	764	184
Browns Ferry 2	764	184
Browns Ferry 3	764	184
Brunswick 1	560	184
Brunswick 2	560	184
Clinton	624	183
Columbia Generating Station	764	183
Cooper	548	184
Duane Arnold	368	184
Fermi 2	764	184
FitzPatrick	560	184
Grand Gulf 1	800	183
Hatch 1	560	184
Hatch 2	560	184
Hope Creek 1	764	183
La Salle 1	764	183
La Salle 2	764	183
Limerick 1	764	184
Limerick 2	764	183

Nine Mile Point 2	764	183
Peach Bottom 2	764	184
Peach Bottom 3	764	183
Perry 1	748	183
River Bend 1	624	183
Susquehanna 1	764	184
Susquehanna 2	764	184
Vermont Yankee	368	184
W14x14: 6 units avg.	**121**	**394**
Ginna	121	394
Kewaunee	121	394
Point Beach 1	121	394
Point Beach 2	121	394
Prairie Island 1	121	394
Prairie Island 2	121	394
W15x15: 15 units avg.	**177**	**459**
Arkansas Nuclear 1	177	464
Crystal River 3	177	464
D.C. Cook 1	193	454
Davis-Besse	177	464
Indian Point 2	193	454
Indian Point 3	193	454
Oconee 1	177	464
Oconee 2	177	464
Oconee 3	177	464
Robinson 2	157	454
Surry 1	157	454
Surry 2	157	454
Three Mile Island 1	177	464
Turkey Point 3	157	454
Turkey Point 4	157	454
W17x17: 34 units avg.	**193**	**432**
Beaver Valley 1	157	425
Beaver Valley 2	157	425
Braidwood 1	193	425

Braidwood 2	193	425
Byron 1	193	425
Byron 2	193	425
Callaway	193	425
Catawba 1	193	425
Catawba 2	193	425
Comanche Peak 1	193	425
Comanche Peak 2	193	425
D.C. Cook 2	193	425
Diablo Canyon 1	193	425
Diablo Canyon 2	193	425
Farley 1	157	425
Farley 2	157	425
Harris 1	157	425
McGuire 1	193	425
McGuire 2	193	425
Millstone 3	193	425
North Anna 1	157	425
North Anna 2	157	425
Salem 1	193	425
Salem 2	193	425
Seabrook 1	193	425
Sequoyah 1	193	425
Sequoyah 2	193	425
South Texas 1	193	542
South Texas 2	193	542
Summer	157	425
Vogtle 1	193	425
Vogtle 2	193	425
Watts Bar 1	193	425
Wolf Creek 1	193	425

TABLE X. Capacity factors for U.S. reactors from 2003 to 2010.

Name of Reactor	**2003**	**2004**	**2005**	**2006**	**2007**	**2008**	**2009**	**2010**
Arkansas Nuclear One, Unit 1	92%	92%	78%	102%	94%	83%	99%	90%
Arkansas Nuclear One, Unit 2	90%	99%	91%	90%	99%	91%	90%	97%
Beaver Valley Power Station, Unit 1	83%	93%	101%	78%	95%	101%	92%	91%

Beaver Valley Power Station, Unit 2	91%	100%	93%	87%	103%	87%	84%	102%
Braidwood Station, Unit 1	97%	95%	100%	96%	92%	101%	95%	89%
Braidwood Station, Unit 2	96%	101%	94%	95%	100%	92%	93%	99%
Browns Ferry Nuclear Plant, Unit 1					49%	88%	94%	86%
Browns Ferry Nuclear Plant, Unit 2	86%	100%	90%	94%	78%	98%	94%	91%
Browns Ferry Nuclear Plant, Unit 3	96%	89%	94%	89%	93%	81%	95%	81%
Brunswick Steam Electric Plant, Unit 1	101%	86%	94%	87%	96%	85%	98%	83%
Brunswick Steam Electric Plant, Unit 2	99%	98%	86%	90%	87%	95%	80%	99%
Byron Station, Unit 1	94%	102%	94%	91%	98%	95%	94%	101%
Byron Station, Unit 2	101%	96%	96%	102%	89%	96%	102%	96%
Callaway Plant	97%	78%	77%	97%	90%	90%	98%	86%
Calvert Cliffs Nuclear Power Plant, Unit 1	102%	92%	100%	84%	99%	93%	98%	90%
Calvert Cliffs Nuclear Power Plant, Unit 2	82%	100%	94%	98%	90%	99%	93%	97%
Catawba Nuclear Station, Unit 1	83%	98%	93%	82%	102%	89%	91%	100%
Catawba Nuclear Station, Unit 2	94%	89%	102%	89%	84%	103%	90%	92%
Clinton Power Station, Unit 1	97%	88%	94%	90%	101%	99%	97%	92%
Columbia Generating Station, Unit 2	79%	91%	83%	94%	82%	93%	67%	95%
Comanche Peak Steam Electric Station, Unit 1	101%	90%	92%	102%	85%	96%	100%	91%
Comanche Peak Steam Electric Station, Unit 2	83%	99%	92%	95%	102%	95%	94%	104%
Cooper Nuclear Station	68%	93%	89%	89%	100%	90%	72%	100%
Crystal River Nuclear Generating Plant, Unit 3	90%	99%	87%	95%	91%	95%	95%	0%
Davis-Besse Nuclear Power Station, Unit 1	-1%	75%	94%	82%	99%	97%	99%	66%
Diablo Canyon Nuclear Power Plant, Unit 1	101%	76%	87%	101%	90%	98%	84%	88%
Diablo Canyon Nuclear Power Plant, Unit 2	81%	84%	99%	87%	99%	74%	84%	100%
Donald C. Cook Nuclear Power Plant, Unit 1	74%	99%	91%	81%	103%	64%	3%	88%
Donald C. Cook Nuclear Power Plant, Unit 2	75%	84%	100%	89%	86%	101%	87%	84%
Dresden Nuclear Power Station, Unit 2	90%	78%	87%	96%	92%	98%	91%	102%
Dresden Nuclear Power Station, Unit 3	94%	85%	93%	94%	100%	93%	97%	90%
Duane Arnold Energy Center	81%	100%	89%	100%	89%	103%	92%	89%
Edwin I. Hatch Nuclear Plant, Unit 1	95%	90%	91%	84%	98%	84%	94%	85%
Edwin I. Hatch Nuclear Plant, Unit 2	90%	97%	87%	99%	87%	96%	67%	96%
Fermi, Unit 2	83%	87%	90%	76%	85%	98%	75%	80%
Fort Calhoun Station, Unit 1	84%	97%	70%	74%	104%	83%	100%	102%
Grand Gulf Nuclear Station, Unit 1	99%	92%	91%	94%	84%	86%	100%	88%
H. B. Robinson Steam Electric Plant, Unit 2	104%	92%	93%	104%	92%	87%	104%	57%
Hope Creek Generating Station, Unit 1	79%	65%	83%	92%	87%	108%	95%	93%
Indian Point Nuclear Generating, Unit 2	99%	88%	99%	89%	99%	91%	98%	82%
Indian Point Nuclear Generating, Unit 3	88%	101%	90%	100%	87%	107%	85%	99%

James A. FitzPatrick Nuclear Power Plant	96%	87%	95%	91%	93%	89%	99%	85%
Joseph M. Farley Nuclear Plant, Unit 1	91%	86%	99%	86%	88%	97%	90%	88%
Joseph M. Farley Nuclear Plant, Unit 2	100%	89%	84%	101%	87%	90%	96%	88%
Kewaunee Power Station	88%	79%	63%	75%	95%	90%	93%	102%
LaSalle County Station, Unit 1	92%	92%	100%	93%	99%	100%	99%	94%
LaSalle County Station, Unit 2	91%	101%	91%	102%	95%	94%	93%	101%
Limerick Generating Station, Unit 1	101%	95%	99%	93%	101%	95%	101%	91%
Limerick Generating Station, Unit 2	94%	99%	91%	100%	91%	101%	94%	99%
McGuire Nuclear Station, Unit 1	103%	85%	93%	103%	79%	87%	104%	92%
McGuire Nuclear Station, Unit 2	94%	103%	89%	87%	103%	90%	94%	104%
Millstone Power Station, Unit 2	80%	98%	88%	84%	100%	86%	81%	97%
Millstone Power Station, Unit 3	101%	88%	86%	100%	86%	88%	105%	86%
Monticello Nuclear Generating Plant, Unit 1	92%	101%	89%	101%	84%	97%	83%	94%
Nine Mile Point Nuclear Station, Unit 1	80%	92%	85%	98%	88%	98%	92%	97%
Nine Mile Point Nuclear Station, Unit 2	96%	86%	100%	90%	92%	90%	99%	89%
North Anna Power Station, Unit 1	81%	91%	95%	88%	89%	101%	92%	86%
North Anna Power Station, Unit 2	90%	92%	87%	100%	85%	82%	100%	100%
Oconee Nuclear Station, Unit 1	71%	98%	91%	79%	99%	84%	85%	100%
Oconee Nuclear Station, Unit 2	102%	76%	90%	100%	91%	86%	103%	91%
Oconee Nuclear Station, Unit 3	85%	77%	98%	91%	87%	102%	94%	91%
Oyster Creek Nuclear Generating Station, Unit 1	97%	89%	99%	86%	94%	83%	92%	85%
Palisades Nuclear Plant	95%	92%	79%	98%	86%	99%	90%	92%
Palo Verde Nuclear Generating Station, Unit 1	97%	85%	63%	42%	77%	86%	101%	81%
Palo Verde Nuclear Generating Station, Unit 2	72%	92%	82%	85%	95%	74%	83%	101%
Palo Verde Nuclear Generating Station, Unit 3	88%	75%	84%	86%	64%	97%	83%	89%
Peach Bottom Atomic Power Station, Unit 2	95%	91%	98%	93%	101%	89%	102%	92%
Peach Bottom Atomic Power Station, Unit 3	91%	102%	91%	102%	93%	99%	89%	100%
Perry Nuclear Power Plant, Unit 1	79%	94%	71%	97%	75%	98%	67%	98%
Pilgrim Nuclear Power Station	83%	99%	91%	97%	85%	97%	90%	99%
Point Beach Nuclear Plant, Unit 1	97%	81%	81%	100%	85%	87%	98%	88%
Point Beach Nuclear Plant, Unit 2	83%	97%	72%	91%	99%	89%	84%	96%
Prairie Island Nuclear Generating Plant, Unit 1	101%	79%	99%	85%	92%	84%	97%	96%
Prairie Island Nuclear Generating Plant, Unit 2	93%	102%	84%	84%	93%	85%	97%	86%
Quad Cities Nuclear Power Station, Unit 1	90%	85%	83%	89%	92%	96%	82%	99%
Quad Cities Nuclear Power Station, Unit 2	92%	81%	93%	86%	99%	86%	91%	92%
River Bend Station, Unit 1	89%	87%	93%	88%	85%	82%	113%	98%
R.E. Ginna Nuclear Power Plant	89%	99%	92%	95%	113%	109%	91%	97%
St. Lucie Plant, Unit 1	102%	86%	83%	102%	85%	91%	100%	72%

St. Lucie Plant, Unit 2	80%	92%	86%	82%	70%	99%	80%	100%
Salem Nuclear Generating Station, Unit 1	94%	72%	92%	99%	89%	91%	99%	85%
Salem Nuclear Generating Station, Unit 2	82%	88%	90%	92%	98%	83%	93%	98%
San Onofre Nuclear Generating Station, Unit 2	104%	86%	95%	72%	89%	91%	60%	75%
San Onofre Nuclear Generating Station, Unit 3	91%	74%	100%	72%	94%	69%	104%	72%
Seabrook Station, Unit 1	91%	100%	89%	86%	99%	89%	81%	100%
Sequoyah Nuclear Plant, Unit 1	73%	92%	100%	90%	87%	101%	89%	84%
Sequoyah Nuclear Plant, Unit 2	84%	96%	90%	90%	100%	89%	89%	97%
Shearon Harris Nuclear Power Plant, Unit 1	92%	89%	101%	89%	94%	99%	94%	90%
South Texas Project, Unit 1	61%	99%	88%	91%	105%	95%	90%	101%
South Texas Project, Unit 2	79%	92%	89%	100%	93%	95%	101%	88%
Surry Nuclear Power Station, Unit 1	76%	92%	96%	90%	89%	98%	94%	89%
Surry Nuclear Power Station, Unit 2	79%	101%	93%	88%	101%	94%	92%	100%
Susquehanna Steam Electric Station, Unit 1	96%	80%	95%	86%	95%	89%	101%	80%
Susquehanna Steam Electric Station, Unit 2	86%	100%	89%	93%	88%	100%	90%	96%
Three Mile Island Nuclear Station, Unit 1	90%	102%	98%	105%	97%	107%	86%	94%
Turkey Point Nuclear Generating, Unit 3	90%	78%	96%	92%	97%	101%	86%	88%
Turkey Point Nuclear Generating, Unit 4		70%	89%	100%	86%	89%	99%	98%
Vermont Yankee Nuclear Power Plant, Unit 1	100%	87%	92%	115%	87%	89%	99%	88%
Virgil C. Summer Nuclear Station, Unit 1	87%	97%	88%	89%	85%	87%	81%	100%
Vogtle Electric Generating Plant, Unit 1	93%	100%	91%	86%	99%	93%	91%	102%
Vogtle Electric Generating Plant, Unit 2	97%	91%	85%	92%	83%	88%	101%	93%
Waterford Steam Electric Station, Unit 3	89%	101%	78%	92%	98%	89%	87%	100%
Watts Bar Nuclear Plant, Unit 1	87%	100%	90%	68%	102%	82%	94%	99%
Wolf Creek Generating Station, Unit 1	87%	99%	86%	92%	102%	83%	86%	86%

APPENDIX B

Appendix B contains information that was relevant during the analysis discussed in Section 4. TABLE XI through TABLE XX list the assembly designs used by each reactor subtype, along with their average enrichments and initial uranium loadings.[20] The tables also list which reactor units belong to each assembly class. TABLE XXI presents the averaged nuclide masses discussed in subsection 4.2, and TABLE XXII and TABLE XXIII present the numerical scores of each nuclide as discussed in subsection 4.4.

TABLE XI. Assembly Designs Used and Reactors Belonging to the B&W 15x15 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
B1515B3	2.42	464	11228.8	452771.2	181		Arkansas 1
B1717B	2.84	457	12978.8	444021.2	4		Davis-Besse
B1515B4	2.91	464	13502.4	450497.6	4136		Oconee 1
B1515B5	3.13	468	14648.4	453351.6	58		Oconee 2
B1515B5Z	3.22	464	14940.8	449059.2	29		Oconee 3
B1515B6	3.47	462	16031.4	445968.6	130		Three Mile Island 1
B1515BZ	3.47	463	16066.1	446933.9	848		
B1515B7	3.51	463	16251.3	446748.7	96		
B1515B	3.57	463	16529.1	446470.9	103		
B1515BGD	3.92	430	16856	413144	4		
B1515B8	3.65	465	16972.5	448027.5	798		
B1515B4Z	3.84	464	17817.6	446182.4	170		
B1515B9	3.96	464	18374.4	445625.6	276		
B1515B10	3.9	477	18603	458397	681		
B1515W	4.06	462	18757.2	443242.8	8		
Database values	4	464	18560	445440			

TABLE XII. Assembly Designs Used and Reactors Belonging to the CE 14x14 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
C1414C	3.2	382	12224	369776	4198		Calvert Cliffs 1
C1414W	3.15	403	12694.5	390305.5	552		Calvert Cliffs 2
C1414A	3.5	381	13335	367665	1383		Millstone 2
Database values	4	382	15280	366720			Saint Lucie 1

TABLE XIII. Assembly Designs Used and Reactors Belonging to the CE 16x16 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
C1616CSD	3.62	414	14986.8	399013.2	4080		Arkansas 2
Database values	4	416	16640	399360			San Onofre 2
							San Onofre 3
							Waterford 3

TABLE XIV. Assembly Designs Used and Reactors Belonging to the CE 16x16 System 80 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
C8016C	3.57	421	15029.7	405970.3	2747		Palo Verde 1
Database values	4	413	16520	396480			Palo Verde 2
							Palo Verde 3

TABLE XV. Assembly Designs Used and Reactors Belonging to the GE BWR 2,3 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
G2307G2A	2.1	195	4095	190905	1672		Dresden 2
G2307G2B	2.15	193	4149.5	188850.5	5047		Dresden 3
G2307G3	2.41	187	4506.7	182493.3	395		Monticello
G2308G4	2.51	184	4618.4	179381.6	3944		Nine Mile Point 1
G2308A	2.66	175	4655	170345	1517		Oyster Creek
G2308G5	2.66	177	4708.2	172291.8	810		Pilgrim 1
G2307A	2.64	182	4804.8	177195.2	152		Quad Cities 1
G2308AP	2.83	173	4895.9	168104.1	32		Quad Cities 2
G2308GP	2.77	177	4902.9	172097.1	3281		
G2308GB	2.8	178	4984	173016	1367		
G2309A	3.1	168	5208	162792	1820		
G2308G7	2.97	179	5316.3	173683.7	164		
G2308G8A	3.09	176	5438.4	170561.6	1150		
G2308G9	3.18	172	5469.6	166530.4	890		
G2308G8B	3.19	173	5518.7	167481.3	1382		
G2308G10	3.25	172	5590	166410	1404		
G2309AIX	3.31	169	5593.9	163406.1	224		
G2309G11	3.56	166	5909.6	160090.4	132		
Database values	4	180	7200	172800			

TABLE XVI. Assembly Designs Used and Reactors Belonging to the GE BWR 4,6 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
G4607G2	1.56	195	3042	191958	1142		Browns Ferry 1
G4608G5	2.36	183	4318.8	178681.2	4213		Browns Ferry 2
G4608G4B	2.31	187	4319.7	182680.3	560		Browns Ferry 3
G4608GP	2.38	183	4355.4	178644.6	12642		Brunswick 1
G4607G3A	2.33	187	4357.1	182642.9	3752		Brunswick 2
G4607G3B	2.31	190	4389	185611	1184		Clinton
G4608W	2.85	157	4474.5	152525.5	8		Cooper
G4608GB	2.53	185	4680.5	180319.5	10246		Duane Arnold
G4608G4A	2.62	184	4820.8	179179.2	1785		Fermi 2
G4609A9X	2.87	169	4850.3	164149.7	768		FitzPatrick
G4608AP	2.88	176	5068.8	170931.2	1888		Grand Gulf
G4609AX+	3.14	167	5243.8	161756.2	8		Hatch
G4608G8	3.19	180	5742	174258	3919		Hope Creek
G4608G9	3.23	178	5749.4	172250.6	7146		LaSalle 1
G4608G10	3.24	178	5767.2	172232.8	2337		LaSalle 2
G4609A5	3.28	176	5772.8	170227.2	1124		Limerick 1
G4608G11	3.38	171	5779.8	165220.2	236		Limerick 2
G4610C	3.29	176	5790.4	170209.6	148		Nine Mile Point 2
G4609A	3.42	173	5916.6	167083.4	3280		Peach Bottom 2
G4610AIX	3.39	175	5932.5	169067.5	4		Peach Bottom 3
G4609G11	3.56	170	6052	163948	5351		Perry
G4609AIX	3.58	175	6265	168735	12		River Bend
G4609G13	3.85	171	6583.5	164416.5	2060		Susquehanna 1
G4608G12	3.71	181	6715.1	174284.9	88		Susquehanna 2
G4610A	3.94	177	6973.8	170026.2	116		Vermont Yankee
G4610G12	3.98	176	7004.8	168995.2	371		
G4610G14	4.11	179	7356.9	171643.1	5		
Database Values	4	184	7360	176640			

TABLE XVII. Assembly Designs Used and Reactors Belonging to the W 14x14 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
W1414W	3.04	394	11977.6	382022.4	603		Ginna
W1414WL	3.07	399	12249.3	386750.7	1429		Kewaunee
W1414ATR	3.38	363	12269.4	350730.6	288		Point Beach 1
W1414B	3.22	383	12332.6	370667.4	2		Point Beach 2
W1414A	3.42	378	12927.6	365072.4	1018		Prairie Island 1
W1414WO	3.92	356	13955.2	342044.8	2108		Prairie Island 2
Database values	4	394	15760	378240			

TABLE XVIII. Assembly Designs Used and Reactors Belonging to the W 15x15 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
W1515APL	1.55	307	4758.5	302241.5	24		Cook 1
W1515A	3	429	12870	416130	889		Indian Point 2
W1515W	3	451	13530	437470	393		Indian Point 3
W1515WL	2.98	455	13559	441441	4644		Robinson
W1515WO	3.53	461	16273.3	444726.7	3576		Surry 1
W1515AHT	4.08	435	17748	417252	308		Surry 2
W1515WV5	3.92	458	17953.6	440046.4	531		Turkey Point 3
Database values	4	459	18360	440640			Turkey Point 4

TABLE XIX. Assembly Designs Used and Reactors Belonging to the W17x17 Assembly Class.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
W1717WO	3.05	425	12962.5	412037.5	3204		Beaver Valley 1
W1717WL	3.12	461	14383.2	446616.8	10097		Beaver Valley 2
W1717WV5	4.01	424	17002.4	406997.6	4469		Braidwood 1
W1717A	4.19	414	17346.6	396653.4	1210		Braidwood 2
W1717B	3.84	456	17510.4	438489.6	2060		Byron 1
W1717WV+	4.16	424	17638.4	406361.6	2126		Byron 2
W1717WVH	3.87	462	17879.4	444120.6	3868		Callaway
W1717WVJ	3.99	462	18433.8	443566.2	104		Catawba 1
W1717WV	4.38	425	18615	406385	24		Catawba 2
W1717WRF	4.18	455	19019	435981	72		Comanche Peak 1
W1717WP	4.59	417	19140.3	397859.7	216		Comanche Peak 2
Database values	4	432	17280	414720			Cook 2
							Diablo Canyon 1
							Diablo Canyon 2
							Farley 1
							Farley 2
							Harris
							McGuire 1
							McGuire 2
							Millstone 3
							North Anna 1
							North Anna 2
							Salem 1
							Salem 2
							Seabrook
							Sequoyah 1
							Sequoyah 2
							Summer
							Vogtle 1 & 2
							Watts Bar 1
							Wolf Creek 1

TABLE XX. Assembly Designs Used and Reactors Belonging to Individual Assembly Classes.

Assembly Design	Average Enrichment (wt.%)	Initial Loading	^{235}U Mass (g)	^{238}U Mass (g)	Number used through 2004		Reactors
Fort Calhoun							
XFC14C	2.96	362	10715.2	351284.8	418		Fort Calhoun
XFC14A	3.57	353	12602.1	340397.9	192		
XFC14W	3.75	374	14025	359975	229		
Database values	4	416	16640	399360			
Palisades							
XPA15C	2.47	412	10176.4	401823.6	273		Palisades
XPA15A	3.17	397	12584.9	384415.1	808		
Database values	4	416	16640	399360			
Saint Lucie Unit 2							
XSL16C	3.44	381	13106.4	367893.6	909		St. Lucie 2
Database values	4	382	15280	366720			
South Texas							
WST17W	3.38	540	18252	521748	1254		South Texas 1
Database values	4	542	21680	520320			South Texas 2

TABLE XXI. Averaged masses (kg) of nuclides at modeled U-235 enrichment levels, used in initial assessment discussed in Section 4.2.

Nuclide	1.50%	1.60%	1.80%	1.90%	2.10%	2.20%	2.30%	2.35%
Am-239	1.939E-12	2.159E-12	2.356E-12	2.347E-12	2.328E-12	2.318E-12	2.308E-12	2.302E-12
Am-240	8.352E-10	9.321E-10	1.019E-09	1.015E-09	1.007E-09	1.003E-09	9.985E-10	9.962E-10
Am-241	1.832E-02	1.909E-02	2.042E-02	2.040E-02	2.037E-02	2.035E-02	2.032E-02	2.031E-02
Am-242	5.822E-05	6.198E-05	6.471E-05	6.445E-05	6.390E-05	6.360E-05	6.328E-05	6.312E-05
Am-242m	3.555E-04	3.921E-04	4.293E-04	4.289E-04	4.279E-04	4.274E-04	4.269E-04	4.266E-04
Am-243	2.404E-01	2.405E-01	2.357E-01	2.304E-01	2.202E-01	2.151E-01	2.100E-01	2.075E-01
Am-244	2.681E-04	2.770E-04	2.740E-04	2.676E-04	2.550E-04	2.487E-04	2.424E-04	2.392E-04
Am-245	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-246	1.309E-09	1.312E-09	1.222E-09	1.179E-09	1.099E-09	1.059E-09	1.020E-09	1.001E-09
Bk-249	2.407E-13	2.351E-13	2.089E-13	2.014E-13	1.871E-13	1.800E-13	1.731E-13	1.698E-13
Bk-250	2.995E-07	2.968E-07	2.793E-07	2.579E-07	2.203E-07	2.031E-07	1.872E-07	1.797E-07
Bk-251	6.137E-10	5.958E-10	5.353E-10	4.946E-10	4.230E-10	3.902E-10	3.598E-10	3.455E-10
Cd-113m	5.599E-14	5.194E-14	4.248E-14	3.917E-14	3.337E-14	3.071E-14	2.827E-14	2.712E-14
Cf-249	1.287E-06	1.354E-06	1.416E-06	1.399E-06	1.368E-06	1.353E-06	1.338E-06	1.330E-06
Cf-250	4.813E-08	4.353E-08	4.222E-08	3.883E-08	3.288E-08	3.018E-08	2.770E-08	2.654E-08
Cf-251	1.044E-07	9.869E-08	8.915E-08	8.218E-08	6.994E-08	6.436E-08	5.919E-08	5.679E-08
Cf-252	6.810E-08	6.714E-08	6.412E-08	5.900E-08	5.004E-08	4.596E-08	4.220E-08	4.045E-08
Cf-253	1.325E-07	1.228E-07	1.052E-07	9.599E-08	8.006E-08	7.292E-08	6.639E-08	6.337E-08
Cf-254	3.659E-10	3.401E-10	2.813E-10	2.567E-10	2.142E-10	1.951E-10	1.777E-10	1.696E-10
Cf-255	4.575E-11	4.105E-11	3.075E-11	2.801E-11	2.328E-11	2.116E-11	1.924E-11	1.834E-11
Cm-241	1.235E-16	1.098E-16	7.724E-17	7.052E-17	5.888E-17	5.363E-17	4.887E-17	4.664E-17
Cm-242	8.390E-10	9.284E-10	1.005E-09	9.967E-10	9.788E-10	9.692E-10	9.594E-10	9.544E-10
Cm-243	1.051E-02	1.117E-02	1.165E-02	1.157E-02	1.141E-02	1.132E-02	1.123E-02	1.118E-02
Cm-244	4.771E-04	5.100E-04	5.414E-04	5.351E-04	5.224E-04	5.157E-04	5.089E-04	5.055E-04
Cm-245	1.940E-01	1.942E-01	1.883E-01	1.815E-01	1.687E-01	1.624E-01	1.563E-01	1.534E-01
Cm-246	1.076E-02	1.144E-02	1.198E-02	1.150E-02	1.060E-02	1.016E-02	9.742E-03	9.538E-03
Cm-247	3.735E-03	3.648E-03	3.446E-03	3.259E-03	2.917E-03	2.754E-03	2.600E-03	2.527E-03
Cm-248	9.478E-05	9.291E-05	8.780E-05	8.230E-05	7.237E-05	6.772E-05	6.336E-05	6.128E-05
Cm-249	1.760E-05	1.682E-05	1.527E-05	1.414E-05	1.216E-05	1.125E-05	1.040E-05	1.000E-05
Cm-250	3.243E-10	3.202E-10	2.950E-10	2.735E-10	2.353E-10	2.177E-10	2.014E-10	1.937E-10
Cm-251	2.916E-11	2.766E-11	2.507E-11	2.300E-11	1.940E-11	1.777E-11	1.628E-11	1.558E-11

TABLE XXI. Continued. (Am-239 to Cm-251; 2.40% to 3.00%)

Nuclide	2.40%	2.50%	2.55%	2.60%	2.70%	2.80%	2.90%	3.00%
Am-239	2.297E-12	2.285E-12	2.279E-12	2.273E-12	2.260E-12	2.247E-12	2.233E-12	2.219E-12
Am-240	9.938E-10	9.888E-10	9.863E-10	9.837E-10	9.783E-10	9.727E-10	9.668E-10	9.608E-10
Am-241	2.030E-02	2.027E-02	2.026E-02	2.025E-02	2.022E-02	2.019E-02	2.015E-02	2.012E-02
Am-242	6.295E-05	6.261E-05	6.244E-05	6.226E-05	6.188E-05	6.150E-05	6.109E-05	6.067E-05
Am-242m	4.263E-04	4.257E-04	4.253E-04	4.250E-04	4.243E-04	4.235E-04	4.227E-04	4.219E-04
Am-243	2.050E-01	2.000E-01	1.975E-01	1.951E-01	1.901E-01	1.853E-01	1.805E-01	1.758E-01
Am-244	2.361E-04	2.300E-04	2.268E-04	2.238E-04	2.176E-04	2.116E-04	2.056E-04	1.996E-04
Am-245	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-246	9.823E-10	9.457E-10	9.271E-10	9.094E-10	8.738E-10	8.398E-10	8.064E-10	7.737E-10
Bk-249	1.665E-13	1.600E-13	1.567E-13	1.535E-13	1.473E-13	1.412E-13	1.353E-13	1.295E-13
Bk-250	1.725E-07	1.590E-07	1.524E-07	1.463E-07	1.344E-07	1.236E-07	1.136E-07	1.042E-07
Bk-251	3.318E-10	3.059E-10	2.932E-10	2.815E-10	2.587E-10	2.380E-10	2.187E-10	2.007E-10
Cd-113m	2.601E-14	2.394E-14	2.292E-14	2.199E-14	2.017E-14	1.852E-14	1.698E-14	1.556E-14
Cf-249	1.323E-06	1.309E-06	1.301E-06	1.294E-06	1.280E-06	1.266E-06	1.253E-06	1.239E-06
Cf-250	2.543E-08	2.334E-08	2.232E-08	2.137E-08	1.956E-08	1.792E-08	1.640E-08	1.498E-08
Cf-251	5.447E-08	5.010E-08	4.797E-08	4.600E-08	4.218E-08	3.872E-08	3.550E-08	3.251E-08
Cf-252	3.876E-08	3.559E-08	3.404E-08	3.261E-08	2.985E-08	2.735E-08	2.503E-08	2.287E-08
Cf-253	6.046E-08	5.507E-08	5.244E-08	5.004E-08	4.540E-08	4.126E-08	3.744E-08	3.395E-08
Cf-254	1.618E-10	1.474E-10	1.404E-10	1.339E-10	1.216E-10	1.105E-10	1.002E-10	9.084E-11
Cf-255	1.748E-11	1.589E-11	1.512E-11	1.442E-11	1.306E-11	1.184E-11	1.072E-11	9.694E-12
Cm-241	4.450E-17	4.054E-17	3.861E-17	3.685E-17	3.344E-17	3.038E-17	2.755E-17	2.497E-17
Cm-242	9.493E-10	9.390E-10	9.337E-10	9.284E-10	9.174E-10	9.063E-10	8.947E-10	8.830E-10
Cm-243	1.114E-02	1.104E-02	1.099E-02	1.094E-02	1.084E-02	1.074E-02	1.064E-02	1.053E-02
Cm-244	5.021E-04	4.952E-04	4.916E-04	4.881E-04	4.809E-04	4.737E-04	4.663E-04	4.589E-04
Cm-245	1.505E-01	1.448E-01	1.419E-01	1.392E-01	1.337E-01	1.285E-01	1.234E-01	1.184E-01
Cm-246	9.336E-03	8.944E-03	8.747E-03	8.560E-03	8.185E-03	7.830E-03	7.484E-03	7.148E-03
Cm-247	2.455E-03	2.317E-03	2.249E-03	2.184E-03	2.057E-03	1.939E-03	1.826E-03	1.717E-03
Cm-248	5.927E-05	5.544E-05	5.354E-05	5.177E-05	4.830E-05	4.510E-05	4.207E-05	3.919E-05
Cm-249	9.622E-06	8.897E-06	8.542E-06	8.212E-06	7.573E-06	6.989E-06	6.446E-06	5.936E-06
Cm-250	1.863E-10	1.723E-10	1.654E-10	1.591E-10	1.467E-10	1.354E-10	1.248E-10	1.149E-10
Cm-251	1.490E-11	1.365E-11	1.304E-11	1.248E-11	1.139E-11	1.041E-11	9.499E-12	8.661E-12

TABLE XXI. Continued. (Am-239 to Cm-251; 3.10% to 3.50%)

Nuclide	3.10%	3.15%	3.20%	3.25%	3.30%	3.40%	3.45%	3.50%
Am-239	2.204E-12	2.197E-12	2.189E-12	2.181E-12	2.173E-12	2.157E-12	2.148E-12	2.140E-12
Am-240	9.544E-10	9.513E-10	9.479E-10	9.446E-10	9.412E-10	9.342E-10	9.307E-10	9.270E-10
Am-241	2.008E-02	2.006E-02	2.004E-02	2.002E-02	2.000E-02	1.995E-02	1.993E-02	1.991E-02
Am-242	6.024E-05	6.002E-05	5.979E-05	5.957E-05	5.933E-05	5.885E-05	5.861E-05	5.836E-05
Am-242m	4.210E-04	4.205E-04	4.200E-04	4.195E-04	4.190E-04	4.179E-04	4.174E-04	4.168E-04
Am-243	1.711E-01	1.688E-01	1.665E-01	1.642E-01	1.619E-01	1.575E-01	1.553E-01	1.530E-01
Am-244	1.937E-04	1.908E-04	1.879E-04	1.851E-04	1.821E-04	1.765E-04	1.737E-04	1.709E-04
Am-245	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-246	7.419E-10	7.264E-10	7.112E-10	6.963E-10	6.810E-10	6.520E-10	6.380E-10	6.236E-10
Bk-249	1.239E-13	1.212E-13	1.185E-13	1.159E-13	1.132E-13	1.080E-13	1.056E-13	1.031E-13
Bk-250	9.551E-08	9.148E-08	8.758E-08	8.390E-08	8.019E-08	7.344E-08	7.031E-08	6.715E-08
Bk-251	1.841E-10	1.763E-10	1.688E-10	1.617E-10	1.545E-10	1.415E-10	1.355E-10	1.294E-10
Cd-113m	1.425E-14	1.363E-14	1.303E-14	1.248E-14	1.192E-14	1.089E-14	1.041E-14	9.940E-15
Cf-249	1.226E-06	1.220E-06	1.214E-06	1.207E-06	1.201E-06	1.189E-06	1.182E-06	1.176E-06
Cf-250	1.368E-08	1.308E-08	1.250E-08	1.195E-08	1.140E-08	1.040E-08	9.937E-09	9.475E-09
Cf-251	2.975E-08	2.846E-08	2.722E-08	2.605E-08	2.488E-08	2.274E-08	2.174E-08	2.075E-08
Cf-252	2.089E-08	1.997E-08	1.908E-08	1.824E-08	1.740E-08	1.587E-08	1.516E-08	1.445E-08
Cf-253	3.074E-08	2.926E-08	2.784E-08	2.651E-08	2.518E-08	2.278E-08	2.167E-08	2.057E-08
Cf-254	8.228E-11	7.829E-11	7.448E-11	7.092E-11	6.736E-11	6.089E-11	5.795E-11	5.500E-11
Cf-255	8.763E-12	8.327E-12	7.912E-12	7.525E-12	7.140E-12	6.436E-12	6.117E-12	5.800E-12
Cm-241	2.262E-17	2.151E-17	2.045E-17	1.947E-17	1.849E-17	1.669E-17	1.587E-17	1.506E-17
Cm-242	8.710E-10	8.650E-10	8.588E-10	8.527E-10	8.464E-10	8.338E-10	8.275E-10	8.209E-10
Cm-243	1.042E-02	1.036E-02	1.030E-02	1.025E-02	1.019E-02	1.007E-02	1.001E-02	9.953E-03
Cm-244	4.513E-04	4.476E-04	4.438E-04	4.400E-04	4.361E-04	4.284E-04	4.246E-04	4.207E-04
Cm-245	1.136E-01	1.112E-01	1.089E-01	1.067E-01	1.044E-01	1.001E-01	9.797E-02	9.582E-02
Cm-246	6.823E-03	6.666E-03	6.512E-03	6.362E-03	6.209E-03	5.920E-03	5.781E-03	5.639E-03
Cm-247	1.614E-03	1.566E-03	1.518E-03	1.472E-03	1.425E-03	1.339E-03	1.298E-03	1.256E-03
Cm-248	3.649E-05	3.522E-05	3.398E-05	3.280E-05	3.160E-05	2.940E-05	2.836E-05	2.731E-05
Cm-249	5.462E-06	5.241E-06	5.027E-06	4.825E-06	4.621E-06	4.248E-06	4.075E-06	3.900E-06
Cm-250	1.057E-10	1.014E-10	9.724E-11	9.332E-11	8.934E-11	8.210E-11	7.872E-11	7.533E-11
Cm-251	7.890E-12	7.531E-12	7.185E-12	6.862E-12	6.537E-12	5.946E-12	5.675E-12	5.402E-12

TABLE XXI. Continued. (Am-239 to Cm-251; 3.60% to 4.40%)

Nuclide	3.60%	3.70%	3.80%	3.90%	4.00%	4.10%	4.20%	4.40%
Am-239	2.122E-12	2.105E-12	2.086E-12	2.067E-12	2.048E-12	2.028E-12	2.008E-12	1.966E-12
Am-240	9.196E-10	9.120E-10	9.042E-10	8.960E-10	8.879E-10	8.795E-10	8.708E-10	8.529E-10
Am-241	1.986E-02	1.980E-02	1.975E-02	1.969E-02	1.963E-02	1.957E-02	1.950E-02	1.937E-02
Am-242	5.786E-05	5.734E-05	5.681E-05	5.626E-05	5.571E-05	5.514E-05	5.455E-05	5.336E-05
Am-242m	4.157E-04	4.144E-04	4.132E-04	4.118E-04	4.104E-04	4.089E-04	4.074E-04	4.041E-04
Am-243	1.487E-01	1.444E-01	1.402E-01	1.360E-01	1.320E-01	1.280E-01	1.241E-01	1.165E-01
Am-244	1.654E-04	1.600E-04	1.547E-04	1.494E-04	1.443E-04	1.393E-04	1.343E-04	1.248E-04
Am-245	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-246	5.965E-10	5.698E-10	5.442E-10	5.191E-10	4.954E-10	4.724E-10	4.498E-10	4.074E-10
Bk-249	9.827E-14	9.357E-14	8.910E-14	8.470E-14	8.054E-14	7.654E-14	7.260E-14	6.525E-14
Bk-250	6.146E-08	5.612E-08	5.128E-08	4.676E-08	4.270E-08	3.896E-08	3.546E-08	2.939E-08
Bk-251	1.184E-10	1.081E-10	9.875E-11	9.001E-11	8.214E-11	7.490E-11	6.814E-11	5.638E-11
Cd-113m	9.078E-15	8.269E-15	7.542E-15	6.860E-15	6.246E-15	5.686E-15	5.160E-15	4.251E-15
Cf-249	1.165E-06	1.153E-06	1.142E-06	1.130E-06	1.120E-06	1.109E-06	1.098E-06	1.078E-06
Cf-250	8.640E-09	7.860E-09	7.159E-09	6.505E-09	5.919E-09	5.384E-09	4.885E-09	4.024E-09
Cf-251	1.895E-08	1.727E-08	1.575E-08	1.433E-08	1.306E-08	1.189E-08	1.080E-08	8.913E-09
Cf-252	1.317E-08	1.198E-08	1.090E-08	9.898E-09	8.999E-09	8.177E-09	7.412E-09	6.090E-09
Cf-253	1.860E-08	1.677E-08	1.514E-08	1.363E-08	1.229E-08	1.107E-08	9.949E-09	8.037E-09
Cf-254	4.970E-11	4.478E-11	4.040E-11	3.635E-11	3.276E-11	2.950E-11	2.647E-11	2.135E-11
Cf-255	5.226E-12	4.696E-12	4.226E-12	3.790E-12	3.404E-12	3.056E-12	2.734E-12	2.188E-12
Cm-241	1.359E-17	1.223E-17	1.101E-17	9.887E-18	8.889E-18	7.985E-18	7.148E-18	5.724E-18
Cm-242	8.079E-10	7.947E-10	7.814E-10	7.678E-10	7.543E-10	7.407E-10	7.267E-10	6.988E-10
Cm-243	9.833E-03	9.710E-03	9.587E-03	9.459E-03	9.333E-03	9.206E-03	9.075E-03	8.813E-03
Cm-244	4.130E-04	4.051E-04	3.974E-04	3.895E-04	3.817E-04	3.739E-04	3.660E-04	3.504E-04
Cm-245	9.178E-02	8.780E-02	8.403E-02	8.031E-02	7.681E-02	7.341E-02	7.009E-02	6.387E-02
Cm-246	5.373E-03	5.112E-03	4.865E-03	4.624E-03	4.397E-03	4.180E-03	3.967E-03	3.572E-03
Cm-247	1.179E-03	1.105E-03	1.036E-03	9.693E-04	9.080E-04	8.502E-04	7.947E-04	6.942E-04
Cm-248	2.538E-05	2.354E-05	2.186E-05	2.025E-05	1.878E-05	1.741E-05	1.611E-05	1.379E-05
Cm-249	3.583E-06	3.285E-06	3.014E-06	2.760E-06	2.530E-06	2.318E-06	2.119E-06	1.772E-06
Cm-250	6.916E-11	6.334E-11	5.808E-11	5.313E-11	4.864E-11	4.451E-11	4.065E-11	3.387E-11
Cm-251	4.912E-12	4.454E-12	4.043E-12	3.662E-12	3.321E-12	3.008E-12	2.720E-12	2.222E-12

TABLE XXI. Continued. (Am-239 to Cm-251; 4.50% to 5.00%)

Nuclide	4.50%	4.60%	4.90%	5.00%
Am-239	1.945E-12	1.923E-12	1.856E-12	1.832E-12
Am-240	8.437E-10	8.345E-10	8.055E-10	7.956E-10
Am-241	1.929E-02	1.922E-02	1.897E-02	1.888E-02
Am-242	5.275E-05	5.213E-05	5.021E-05	4.956E-05
Am-242m	4.024E-04	4.006E-04	3.948E-04	3.928E-04
Am-243	1.128E-01	1.093E-01	9.897E-02	9.569E-02
Am-244	1.202E-04	1.157E-04	1.029E-04	9.879E-05
Am-245	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Am-246	3.873E-10	3.682E-10	3.150E-10	2.985E-10
Bk-249	6.177E-14	5.847E-14	4.935E-14	4.656E-14
Bk-250	2.673E-08	2.432E-08	1.823E-08	1.654E-08
Bk-251	5.124E-11	4.658E-11	3.478E-11	3.150E-11
Cd-113m	3.854E-15	3.496E-15	2.589E-15	2.339E-15
Cf-249	1.068E-06	1.059E-06	1.032E-06	1.023E-06
Cf-250	3.648E-09	3.311E-09	2.462E-09	2.228E-09
Cf-251	8.089E-09	7.345E-09	5.470E-09	4.951E-09
Cf-252	5.515E-09	4.996E-09	3.694E-09	3.337E-09
Cf-253	7.216E-09	6.483E-09	4.670E-09	4.181E-09
Cf-254	1.915E-11	1.718E-11	1.231E-11	1.101E-11
Cf-255	1.956E-12	1.748E-12	1.236E-12	1.100E-12
Cm-241	5.119E-18	4.574E-18	3.231E-18	2.873E-18
Cm-242	6.847E-10	6.708E-10	6.285E-10	6.144E-10
Cm-243	8.679E-03	8.548E-03	8.144E-03	8.009E-03
Cm-244	3.427E-04	3.350E-04	3.121E-04	3.046E-04
Cm-245	6.092E-02	5.813E-02	5.034E-02	4.793E-02
Cm-246	3.386E-03	3.211E-03	2.729E-03	2.582E-03
Cm-247	6.482E-04	6.055E-04	4.919E-04	4.583E-04
Cm-248	1.274E-05	1.178E-05	9.278E-06	8.556E-06
Cm-249	1.618E-06	1.479E-06	1.124E-06	1.024E-06
Cm-250	3.089E-11	2.819E-11	2.129E-11	1.936E-11
Cm-251	2.006E-12	1.813E-12	1.327E-12	1.195E-12

TABLE XXI. Continued. (Cs-134 to Pu-242; 1.50% to 2.30%)

Nuclide	1.50%	1.60%	1.80%	1.90%	2.10%	2.20%	2.30%
Cs-134	9.495E-02	9.499E-02	9.411E-02	9.384E-02	9.326E-02	9.295E-02	9.262E-02
Cs-135	1.666E-01	1.751E-01	1.853E-01	1.863E-01	1.885E-01	1.897E-01	1.909E-01
Cs-137	6.875E-01	6.875E-01	6.873E-01	6.873E-01	6.874E-01	6.874E-01	6.875E-01
Eu-154	1.660E-02	1.718E-02	1.773E-02	1.765E-02	1.749E-02	1.741E-02	1.733E-02
Eu-155	4.073E-03	4.089E-03	4.074E-03	4.055E-03	4.018E-03	3.999E-03	3.979E-03
He-4	1.460E-03	1.482E-03	1.497E-03	1.477E-03	1.439E-03	1.420E-03	1.401E-03
I-129	1.032E-01	1.029E-01	1.021E-01	1.015E-01	1.002E-01	9.959E-02	9.898E-02
Kr-85	9.289E-03	9.431E-03	9.697E-03	9.834E-03	1.010E-02	1.024E-02	1.037E-02
Nb-94	5.983E-07	5.891E-07	5.723E-07	5.648E-07	5.500E-07	5.425E-07	5.351E-07
Np-235	7.933E-09	8.433E-09	9.093E-09	9.326E-09	9.746E-09	9.941E-09	1.012E-08
Np-236	6.380E-08	7.047E-08	8.054E-08	8.328E-08	8.846E-08	9.100E-08	9.345E-08
Np-236m	2.774E-09	2.999E-09	3.270E-09	3.378E-09	3.579E-09	3.677E-09	3.770E-09
Np-237	1.396E-01	1.459E-01	1.575E-01	1.631E-01	1.736E-01	1.787E-01	1.837E-01
Np-238	5.964E-04	6.250E-04	6.563E-04	6.778E-04	7.177E-04	7.370E-04	7.554E-04
Np-239	3.816E-02	3.872E-02	3.825E-02	3.811E-02	3.781E-02	3.765E-02	3.748E-02
Np-240	1.757E-06	1.767E-06	1.711E-06	1.700E-06	1.677E-06	1.665E-06	1.652E-06
Np-240m	1.269E-17	1.219E-17	1.127E-17	1.088E-17	1.015E-17	9.792E-18	9.443E-18
Np-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pa-231	1.850E-09	2.193E-09	2.739E-09	2.913E-09	3.279E-09	3.480E-09	3.686E-09
Pa-232	7.192E-12	8.402E-12	1.021E-11	1.079E-11	1.197E-11	1.261E-11	1.326E-11
Pa-233	4.861E-09	5.082E-09	5.491E-09	5.686E-09	6.056E-09	6.238E-09	6.414E-09
Pa-234	9.871E-13	1.056E-12	1.139E-12	1.175E-12	1.241E-12	1.273E-12	1.304E-12
Pa-234m	1.549E-13	1.550E-13	1.551E-13	1.552E-13	1.553E-13	1.554E-13	1.554E-13
Pa-235	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pd-107	2.212E-01	2.179E-01	2.115E-01	2.083E-01	2.022E-01	1.991E-01	1.960E-01
Pm-147	5.229E-02	5.334E-02	5.472E-02	5.500E-02	5.559E-02	5.592E-02	5.624E-02
Pu-236	3.751E-07	4.034E-07	4.440E-07	4.554E-07	4.760E-07	4.857E-07	4.948E-07
Pu-237	2.721E-07	3.049E-07	3.415E-07	3.471E-07	3.568E-07	3.611E-07	3.649E-07
Pu-238	1.054E-01	1.115E-01	1.203E-01	1.225E-01	1.263E-01	1.280E-01	1.296E-01
Pu-239	1.702E+00	1.772E+00	1.850E+00	1.850E+00	1.849E+00	1.849E+00	1.848E+00
Pu-240	1.051E+00	1.044E+00	1.038E+00	1.036E+00	1.033E+00	1.032E+00	1.030E+00
Pu-241	5.880E-01	6.142E-01	6.415E-01	6.398E-01	6.363E-01	6.343E-01	6.323E-01
Pu-242	6.200E-01	5.976E-01	5.656E-01	5.582E-01	5.435E-01	5.359E-01	5.284E-01

TABLE XXI. Continued. (Cs-134 to Pu-242; 2.35% to 2.80%)

Nuclide	2.35%	2.40%	2.50%	2.55%	2.60%	2.70%	2.80%
Cs-134	9.246E-02	9.229E-02	9.194E-02	9.175E-02	9.157E-02	9.119E-02	9.081E-02
Cs-135	1.915E-01	1.921E-01	1.934E-01	1.940E-01	1.947E-01	1.960E-01	1.974E-01
Cs-137	6.875E-01	6.876E-01	6.876E-01	6.877E-01	6.877E-01	6.877E-01	6.878E-01
Eu-154	1.729E-02	1.725E-02	1.717E-02	1.712E-02	1.708E-02	1.700E-02	1.691E-02
Eu-155	3.970E-03	3.960E-03	3.940E-03	3.930E-03	3.920E-03	3.899E-03	3.879E-03
He-4	1.392E-03	1.383E-03	1.366E-03	1.357E-03	1.348E-03	1.331E-03	1.315E-03
I-129	9.867E-02	9.837E-02	9.776E-02	9.745E-02	9.715E-02	9.655E-02	9.596E-02
Kr-85	1.043E-02	1.050E-02	1.063E-02	1.070E-02	1.076E-02	1.089E-02	1.102E-02
Nb-94	5.315E-07	5.278E-07	5.206E-07	5.169E-07	5.133E-07	5.060E-07	4.990E-07
Np-235	1.021E-08	1.029E-08	1.044E-08	1.052E-08	1.059E-08	1.072E-08	1.084E-08
Np-236	9.463E-08	9.580E-08	9.808E-08	9.922E-08	1.003E-07	1.025E-07	1.045E-07
Np-236m	3.815E-09	3.859E-09	3.944E-09	3.986E-09	4.026E-09	4.104E-09	4.178E-09
Np-237	1.862E-01	1.886E-01	1.933E-01	1.957E-01	1.979E-01	2.024E-01	2.067E-01
Np-238	7.642E-04	7.729E-04	7.896E-04	7.979E-04	8.057E-04	8.211E-04	8.355E-04
Np-239	3.739E-02	3.730E-02	3.712E-02	3.703E-02	3.693E-02	3.673E-02	3.653E-02
Np-240	1.645E-06	1.639E-06	1.625E-06	1.618E-06	1.611E-06	1.596E-06	1.581E-06
Np-240m	9.273E-18	9.105E-18	8.777E-18	8.611E-18	8.452E-18	8.135E-18	7.831E-18
Np-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pa-231	3.793E-09	3.903E-09	4.129E-09	4.248E-09	4.368E-09	4.619E-09	4.877E-09
Pa-232	1.359E-11	1.393E-11	1.462E-11	1.498E-11	1.534E-11	1.609E-11	1.684E-11
Pa-233	6.499E-09	6.584E-09	6.748E-09	6.831E-09	6.911E-09	7.068E-09	7.219E-09
Pa-234	1.319E-12	1.333E-12	1.361E-12	1.375E-12	1.388E-12	1.414E-12	1.438E-12
Pa-234m	1.554E-13	1.555E-13	1.555E-13	1.555E-13	1.555E-13	1.556E-13	1.556E-13
Pa-235	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pd-107	1.945E-01	1.930E-01	1.900E-01	1.885E-01	1.870E-01	1.841E-01	1.812E-01
Pm-147	5.641E-02	5.659E-02	5.694E-02	5.713E-02	5.731E-02	5.770E-02	5.809E-02
Pu-236	4.990E-07	5.032E-07	5.110E-07	5.148E-07	5.183E-07	5.251E-07	5.312E-07
Pu-237	3.667E-07	3.683E-07	3.713E-07	3.726E-07	3.739E-07	3.760E-07	3.778E-07
Pu-238	1.304E-01	1.311E-01	1.324E-01	1.330E-01	1.336E-01	1.347E-01	1.357E-01
Pu-239	1.848E+00	1.848E+00	1.848E+00	1.847E+00	1.847E+00	1.847E+00	1.846E+00
Pu-240	1.029E+00	1.028E+00	1.026E+00	1.025E+00	1.024E+00	1.022E+00	1.020E+00
Pu-241	6.313E-01	6.302E-01	6.280E-01	6.268E-01	6.256E-01	6.232E-01	6.207E-01
Pu-242	5.246E-01	5.208E-01	5.132E-01	5.093E-01	5.054E-01	4.976E-01	4.899E-01

TABLE XXI. Continued. (Cs-134 to Pu-242; 2.90% to 3.30%)

Nuclide	2.90%	3.00%	3.10%	3.15%	3.20%	3.25%	3.30%
Cs-134	9.040E-02	8.998E-02	8.955E-02	8.933E-02	8.910E-02	8.888E-02	8.864E-02
Cs-135	1.988E-01	2.002E-01	2.016E-01	2.024E-01	2.031E-01	2.038E-01	2.046E-01
Cs-137	6.880E-01	6.881E-01	6.882E-01	6.883E-01	6.883E-01	6.884E-01	6.884E-01
Eu-154	1.682E-02	1.673E-02	1.664E-02	1.660E-02	1.655E-02	1.651E-02	1.646E-02
Eu-155	3.858E-03	3.837E-03	3.816E-03	3.805E-03	3.794E-03	3.783E-03	3.772E-03
He-4	1.298E-03	1.282E-03	1.266E-03	1.259E-03	1.251E-03	1.244E-03	1.236E-03
I-129	9.538E-02	9.480E-02	9.422E-02	9.394E-02	9.366E-02	9.337E-02	9.309E-02
Kr-85	1.115E-02	1.128E-02	1.140E-02	1.146E-02	1.153E-02	1.159E-02	1.165E-02
Nb-94	4.918E-07	4.847E-07	4.777E-07	4.742E-07	4.707E-07	4.673E-07	4.638E-07
Np-235	1.094E-08	1.104E-08	1.112E-08	1.116E-08	1.119E-08	1.122E-08	1.125E-08
Np-236	1.065E-07	1.084E-07	1.103E-07	1.112E-07	1.120E-07	1.129E-07	1.137E-07
Np-236m	4.247E-09	4.314E-09	4.376E-09	4.405E-09	4.433E-09	4.460E-09	4.487E-09
Np-237	2.109E-01	2.150E-01	2.190E-01	2.209E-01	2.228E-01	2.246E-01	2.265E-01
Np-238	8.491E-04	8.620E-04	8.741E-04	8.797E-04	8.852E-04	8.904E-04	8.956E-04
Np-239	3.632E-02	3.610E-02	3.587E-02	3.576E-02	3.564E-02	3.552E-02	3.540E-02
Np-240	1.565E-06	1.549E-06	1.532E-06	1.524E-06	1.515E-06	1.506E-06	1.497E-06
Np-240m	7.533E-18	7.243E-18	6.959E-18	6.821E-18	6.686E-18	6.554E-18	6.418E-18
Np-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pa-231	5.152E-09	5.436E-09	5.732E-09	5.887E-09	6.045E-09	6.202E-09	6.367E-09
Pa-232	1.763E-11	1.844E-11	1.927E-11	1.969E-11	2.013E-11	2.055E-11	2.100E-11
Pa-233	7.365E-09	7.507E-09	7.645E-09	7.711E-09	7.776E-09	7.839E-09	7.903E-09
Pa-234	1.461E-12	1.482E-12	1.502E-12	1.512E-12	1.521E-12	1.530E-12	1.538E-12
Pa-234m	1.556E-13	1.556E-13	1.557E-13	1.557E-13	1.557E-13	1.557E-13	1.557E-13
Pa-235	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pd-107	1.783E-01	1.754E-01	1.726E-01	1.711E-01	1.697E-01	1.684E-01	1.670E-01
Pm-147	5.850E-02	5.892E-02	5.935E-02	5.957E-02	5.980E-02	6.002E-02	6.025E-02
Pu-236	5.368E-07	5.420E-07	5.466E-07	5.487E-07	5.507E-07	5.525E-07	5.542E-07
Pu-237	3.791E-07	3.802E-07	3.808E-07	3.810E-07	3.810E-07	3.810E-07	3.809E-07
Pu-238	1.365E-01	1.372E-01	1.379E-01	1.381E-01	1.384E-01	1.386E-01	1.388E-01
Pu-239	1.845E+00	1.845E+00	1.844E+00	1.843E+00	1.843E+00	1.842E+00	1.842E+00
Pu-240	1.017E+00	1.015E+00	1.012E+00	1.011E+00	1.010E+00	1.008E+00	1.007E+00
Pu-241	6.180E-01	6.153E-01	6.124E-01	6.109E-01	6.094E-01	6.079E-01	6.064E-01
Pu-242	4.820E-01	4.742E-01	4.663E-01	4.623E-01	4.584E-01	4.545E-01	4.505E-01

TABLE XXI. Continued. (Cs-134 to Pu-242; 3.40% to 3.90%)

Nuclide	3.40%	3.45%	3.50%	3.60%	3.70%	3.80%	3.90%
Cs-134	8.817E-02	8.793E-02	8.768E-02	8.719E-02	8.667E-02	8.615E-02	8.562E-02
Cs-135	2.061E-01	2.069E-01	2.077E-01	2.092E-01	2.109E-01	2.125E-01	2.142E-01
Cs-137	6.885E-01	6.885E-01	6.886E-01	6.887E-01	6.888E-01	6.889E-01	6.891E-01
Eu-154	1.637E-02	1.632E-02	1.627E-02	1.618E-02	1.608E-02	1.598E-02	1.589E-02
Eu-155	3.750E-03	3.739E-03	3.728E-03	3.706E-03	3.683E-03	3.660E-03	3.637E-03
He-4	1.221E-03	1.214E-03	1.207E-03	1.193E-03	1.179E-03	1.166E-03	1.152E-03
I-129	9.254E-02	9.227E-02	9.198E-02	9.145E-02	9.091E-02	9.039E-02	8.986E-02
Kr-85	1.177E-02	1.183E-02	1.189E-02	1.201E-02	1.213E-02	1.225E-02	1.237E-02
Nb-94	4.570E-07	4.536E-07	4.502E-07	4.435E-07	4.368E-07	4.302E-07	4.236E-07
Np-235	1.130E-08	1.132E-08	1.134E-08	1.137E-08	1.139E-08	1.140E-08	1.140E-08
Np-236	1.153E-07	1.161E-07	1.169E-07	1.183E-07	1.197E-07	1.210E-07	1.222E-07
Np-236m	4.536E-09	4.559E-09	4.582E-09	4.623E-09	4.661E-09	4.694E-09	4.723E-09
Np-237	2.300E-01	2.317E-01	2.334E-01	2.366E-01	2.398E-01	2.427E-01	2.456E-01
Np-238	9.050E-04	9.095E-04	9.139E-04	9.217E-04	9.288E-04	9.350E-04	9.405E-04
Np-239	3.515E-02	3.503E-02	3.490E-02	3.464E-02	3.437E-02	3.411E-02	3.382E-02
Np-240	1.479E-06	1.470E-06	1.461E-06	1.443E-06	1.423E-06	1.404E-06	1.384E-06
Np-240m	6.161E-18	6.036E-18	5.909E-18	5.668E-18	5.431E-18	5.205E-18	4.983E-18
Np-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pa-231	6.704E-09	6.876E-09	7.056E-09	7.417E-09	7.798E-09	8.187E-09	8.601E-09
Pa-232	2.189E-11	2.234E-11	2.281E-11	2.374E-11	2.470E-11	2.566E-11	2.667E-11
Pa-233	8.025E-09	8.083E-09	8.142E-09	8.253E-09	8.361E-09	8.462E-09	8.560E-09
Pa-234	1.554E-12	1.561E-12	1.569E-12	1.582E-12	1.593E-12	1.604E-12	1.613E-12
Pa-234m	1.557E-13	1.557E-13	1.557E-13	1.556E-13	1.556E-13	1.556E-13	1.556E-13
Pa-235	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pd-107	1.642E-01	1.629E-01	1.615E-01	1.588E-01	1.561E-01	1.536E-01	1.509E-01
Pm-147	6.072E-02	6.096E-02	6.120E-02	6.169E-02	6.220E-02	6.271E-02	6.324E-02
Pu-236	5.573E-07	5.586E-07	5.598E-07	5.619E-07	5.635E-07	5.647E-07	5.654E-07
Pu-237	3.805E-07	3.801E-07	3.797E-07	3.786E-07	3.771E-07	3.754E-07	3.733E-07
Pu-238	1.390E-01	1.391E-01	1.392E-01	1.393E-01	1.393E-01	1.392E-01	1.390E-01
Pu-239	1.841E+00	1.841E+00	1.840E+00	1.839E+00	1.838E+00	1.837E+00	1.836E+00
Pu-240	1.004E+00	1.002E+00	1.001E+00	9.979E-01	9.946E-01	9.912E-01	9.877E-01
Pu-241	6.032E-01	6.015E-01	5.999E-01	5.965E-01	5.930E-01	5.894E-01	5.857E-01
Pu-242	4.426E-01	4.387E-01	4.347E-01	4.268E-01	4.189E-01	4.112E-01	4.033E-01

TABLE XXI. Continued. (Cs-134 to Pu-242; 4.00% to 5.00%)

Nuclide	4.00%	4.10%	4.20%	4.40%	4.50%	4.60%	4.90%	5.00%
Cs-134	8.507E-02	8.452E-02	8.395E-02	8.279E-02	8.219E-02	8.160E-02	7.975E-02	7.911E-02
Cs-135	2.158E-01	2.175E-01	2.192E-01	2.227E-01	2.245E-01	2.263E-01	2.318E-01	2.337E-01
Cs-137	6.892E-01	6.893E-01	6.894E-01	6.896E-01	6.898E-01	6.899E-01	6.901E-01	6.903E-01
Eu-154	1.579E-02	1.569E-02	1.559E-02	1.539E-02	1.528E-02	1.518E-02	1.487E-02	1.477E-02
Eu-155	3.614E-03	3.591E-03	3.567E-03	3.520E-03	3.496E-03	3.473E-03	3.400E-03	3.376E-03
He-4	1.139E-03	1.127E-03	1.115E-03	1.091E-03	1.079E-03	1.068E-03	1.036E-03	1.026E-03
I-129	8.935E-02	8.884E-02	8.834E-02	8.735E-02	8.687E-02	8.640E-02	8.502E-02	8.458E-02
Kr-85	1.248E-02	1.259E-02	1.270E-02	1.292E-02	1.303E-02	1.314E-02	1.344E-02	1.354E-02
Nb-94	4.172E-07	4.108E-07	4.044E-07	3.920E-07	3.858E-07	3.798E-07	3.622E-07	3.564E-07
Np-235	1.139E-08	1.137E-08	1.135E-08	1.127E-08	1.122E-08	1.116E-08	1.095E-08	1.087E-08
Np-236	1.233E-07	1.244E-07	1.254E-07	1.271E-07	1.279E-07	1.286E-07	1.302E-07	1.307E-07
Np-236m	4.748E-09	4.769E-09	4.786E-09	4.809E-09	4.815E-09	4.817E-09	4.802E-09	4.791E-09
Np-237	2.483E-01	2.508E-01	2.533E-01	2.578E-01	2.598E-01	2.617E-01	2.667E-01	2.682E-01
Np-238	9.451E-04	9.489E-04	9.520E-04	9.559E-04	9.569E-04	9.570E-04	9.534E-04	9.509E-04
Np-239	3.354E-02	3.326E-02	3.296E-02	3.237E-02	3.206E-02	3.175E-02	3.080E-02	3.048E-02
Np-240	1.364E-06	1.344E-06	1.323E-06	1.281E-06	1.260E-06	1.239E-06	1.174E-06	1.153E-06
Np-240m	4.771E-18	4.567E-18	4.366E-18	3.989E-18	3.809E-18	3.638E-18	3.160E-18	3.012E-18
Np-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pa-231	9.021E-09	9.454E-09	9.914E-09	1.086E-08	1.136E-08	1.188E-08	1.352E-08	1.409E-08
Pa-232	2.767E-11	2.868E-11	2.974E-11	3.186E-11	3.295E-11	3.403E-11	3.739E-11	3.852E-11
Pa-233	8.651E-09	8.739E-09	8.822E-09	8.974E-09	9.042E-09	9.106E-09	9.271E-09	9.318E-09
Pa-234	1.620E-12	1.627E-12	1.632E-12	1.638E-12	1.640E-12	1.640E-12	1.633E-12	1.629E-12
Pa-234m	1.555E-13	1.555E-13	1.555E-13	1.554E-13	1.553E-13	1.552E-13	1.550E-13	1.549E-13
Pa-235	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pd-107	1.484E-01	1.459E-01	1.434E-01	1.385E-01	1.362E-01	1.338E-01	1.270E-01	1.248E-01
Pm-147	6.377E-02	6.431E-02	6.487E-02	6.600E-02	6.658E-02	6.716E-02	6.897E-02	6.959E-02
Pu-236	5.657E-07	5.656E-07	5.651E-07	5.629E-07	5.613E-07	5.593E-07	5.515E-07	5.483E-07
Pu-237	3.711E-07	3.685E-07	3.656E-07	3.592E-07	3.556E-07	3.519E-07	3.396E-07	3.352E-07
Pu-238	1.387E-01	1.383E-01	1.378E-01	1.367E-01	1.360E-01	1.353E-01	1.326E-01	1.316E-01
Pu-239	1.834E+0	1.833E+0	1.832E+0	1.829E+0	1.827E+0	1.825E+0	1.820E+0	1.818E+0
Pu-240	9.843E-01	9.806E-01	9.767E-01	9.689E-01	9.647E-01	9.606E-01	9.473E-01	9.427E-01
Pu-241	5.820E-01	5.781E-01	5.741E-01	5.658E-01	5.616E-01	5.573E-01	5.438E-01	5.392E-01
Pu-242	3.956E-01	3.879E-01	3.801E-01	3.649E-01	3.573E-01	3.499E-01	3.279E-01	3.207E-01

TABLE XXI. Continued. (Pu-243 to Zr-93; 1.50% to 2.30%)

Nuclide	1.50%	1.60%	1.80%	1.90%	2.10%	2.20%	2.30%
Pu-243	2.025E-04	2.085E-04	2.070E-04	2.039E-04	1.977E-04	1.945E-04	1.912E-04
Pu-244	7.527E-05	7.230E-05	6.685E-05	6.456E-05	6.022E-05	5.809E-05	5.602E-05
Pu-245	6.703E-09	6.719E-09	6.258E-09	6.041E-09	5.627E-09	5.422E-09	5.223E-09
Pu-246	9.634E-11	9.409E-11	8.361E-11	8.060E-11	7.488E-11	7.205E-11	6.929E-11
Ru-106	1.081E-01	1.080E-01	1.057E-01	1.048E-01	1.030E-01	1.020E-01	1.010E-01
Sb-125	4.793E-03	4.775E-03	4.702E-03	4.672E-03	4.611E-03	4.580E-03	4.550E-03
Se-79	2.348E-03	2.360E-03	2.383E-03	2.393E-03	2.413E-03	2.424E-03	2.434E-03
Sm-151	5.829E-03	6.178E-03	6.543E-03	6.545E-03	6.549E-03	6.552E-03	6.555E-03
Sn-126	1.405E-02	1.391E-02	1.365E-02	1.353E-02	1.331E-02	1.320E-02	1.308E-02
Sr-90	1.984E-01	2.018E-01	2.084E-01	2.118E-01	2.184E-01	2.217E-01	2.250E-01
Tc-99	3.894E-01	3.910E-01	3.943E-01	3.952E-01	3.971E-01	3.980E-01	3.990E-01
Th-226	1.275E-18	1.435E-18	1.614E-18	1.641E-18	1.686E-18	1.706E-18	1.723E-18
Th-227	2.136E-15	2.450E-15	2.806E-15	2.849E-15	2.923E-15	2.955E-15	2.985E-15
Th-228	7.582E-10	8.000E-10	8.832E-10	8.966E-10	9.196E-10	9.296E-10	9.387E-10
Th-229	3.088E-10	3.390E-10	3.770E-10	3.803E-10	3.852E-10	3.869E-10	3.882E-10
Th-230	4.081E-09	4.302E-09	4.718E-09	4.790E-09	4.920E-09	4.980E-09	5.038E-09
Th-231	6.287E-12	7.137E-12	8.185E-12	8.410E-12	8.854E-12	9.083E-12	9.312E-12
Th-232	7.098E-08	7.525E-08	8.417E-08	8.841E-08	9.653E-08	1.006E-07	1.046E-07
Th-233	1.883E-13	2.050E-13	2.287E-13	2.395E-13	2.600E-13	2.700E-13	2.797E-13
Th-234	4.511E-09	4.507E-09	4.503E-09	4.502E-09	4.500E-09	4.499E-09	4.497E-09
U-230	1.240E-15	1.395E-15	1.569E-15	1.596E-15	1.640E-15	1.659E-15	1.676E-15
U-231	7.283E-14	8.200E-14	9.233E-14	9.394E-14	9.665E-14	9.783E-14	9.887E-14
U-232	1.100E-07	1.172E-07	1.294E-07	1.319E-07	1.362E-07	1.381E-07	1.399E-07
U-233	7.752E-08	8.636E-08	1.013E-07	1.056E-07	1.142E-07	1.187E-07	1.231E-07
U-234	1.181E-03	1.249E-03	1.363E-03	1.383E-03	1.418E-03	1.434E-03	1.449E-03
U-235	1.976E-01	2.485E-01	3.349E-01	3.682E-01	4.395E-01	4.792E-01	5.204E-01
U-236	6.338E-01	6.955E-01	8.094E-01	8.567E-01	9.497E-01	9.973E-01	1.044E+00
U-237	2.225E-03	2.366E-03	2.554E-03	2.658E-03	2.859E-03	2.960E-03	3.059E-03
U-238	3.114E+02	3.112E+02	3.109E+02	3.109E+02	3.107E+02	3.106E+02	3.105E+02
U-239	2.647E-04	2.686E-04	2.653E-04	2.643E-04	2.623E-04	2.611E-04	2.600E-04
U-240	1.487E-15	1.428E-15	1.320E-15	1.275E-15	1.190E-15	1.147E-15	1.106E-15
U-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zr-93	3.171E-01	3.197E-01	3.251E-01	3.279E-01	3.334E-01	3.362E-01	3.389E-01

TABLE XXI. Continued. (Pu-243 to Zr-93; 2.35% to 2.80%)

Nuclide	2.35%	2.40%	2.50%	2.55%	2.60%	2.70%	2.80%
Pu-243	1.896E-04	1.880E-04	1.847E-04	1.830E-04	1.814E-04	1.780E-04	1.747E-04
Pu-244	5.501E-05	5.401E-05	5.207E-05	5.108E-05	5.014E-05	4.826E-05	4.646E-05
Pu-245	5.127E-09	5.031E-09	4.844E-09	4.748E-09	4.658E-09	4.476E-09	4.302E-09
Pu-246	6.796E-11	6.663E-11	6.404E-11	6.272E-11	6.146E-11	5.894E-11	5.653E-11
Ru-106	1.006E-01	1.001E-01	9.909E-02	9.858E-02	9.809E-02	9.706E-02	9.605E-02
Sb-125	4.535E-03	4.519E-03	4.489E-03	4.473E-03	4.458E-03	4.427E-03	4.397E-03
Se-79	2.439E-03	2.444E-03	2.454E-03	2.459E-03	2.464E-03	2.474E-03	2.484E-03
Sm-151	6.556E-03	6.558E-03	6.562E-03	6.565E-03	6.566E-03	6.570E-03	6.575E-03
Sn-126	1.303E-02	1.297E-02	1.287E-02	1.281E-02	1.276E-02	1.265E-02	1.254E-02
Sr-90	2.266E-01	2.283E-01	2.315E-01	2.331E-01	2.347E-01	2.379E-01	2.410E-01
Tc-99	3.995E-01	4.000E-01	4.010E-01	4.016E-01	4.021E-01	4.031E-01	4.042E-01
Th-226	1.731E-18	1.738E-18	1.750E-18	1.756E-18	1.760E-18	1.769E-18	1.775E-18
Th-227	2.998E-15	3.011E-15	3.034E-15	3.045E-15	3.055E-15	3.074E-15	3.090E-15
Th-228	9.428E-10	9.467E-10	9.538E-10	9.572E-10	9.602E-10	9.657E-10	9.704E-10
Th-229	3.887E-10	3.890E-10	3.895E-10	3.895E-10	3.895E-10	3.892E-10	3.885E-10
Th-230	5.065E-09	5.091E-09	5.142E-09	5.167E-09	5.191E-09	5.237E-09	5.281E-09
Th-231	9.428E-12	9.544E-12	9.779E-12	9.900E-12	1.002E-11	1.027E-11	1.052E-11
Th-232	1.065E-07	1.085E-07	1.123E-07	1.142E-07	1.161E-07	1.198E-07	1.235E-07
Th-233	2.844E-13	2.891E-13	2.981E-13	3.027E-13	3.070E-13	3.156E-13	3.238E-13
Th-234	4.497E-09	4.496E-09	4.495E-09	4.494E-09	4.494E-09	4.492E-09	4.491E-09
U-230	1.683E-15	1.690E-15	1.702E-15	1.708E-15	1.712E-15	1.720E-15	1.726E-15
U-231	9.933E-14	9.975E-14	1.005E-13	1.009E-13	1.012E-13	1.017E-13	1.021E-13
U-232	1.407E-07	1.415E-07	1.429E-07	1.436E-07	1.443E-07	1.455E-07	1.465E-07
U-233	1.253E-07	1.276E-07	1.321E-07	1.344E-07	1.366E-07	1.413E-07	1.459E-07
U-234	1.455E-03	1.462E-03	1.474E-03	1.480E-03	1.486E-03	1.496E-03	1.505E-03
U-235	5.420E-01	5.641E-01	6.099E-01	6.341E-01	6.586E-01	7.101E-01	7.632E-01
U-236	1.068E+0	1.091E+0	1.138E+0	1.162E+0	1.185E+0	1.232E+0	1.278E+0
U-237	3.108E-03	3.156E-03	3.251E-03	3.299E-03	3.345E-03	3.438E-03	3.528E-03
U-238	3.105E+02	3.105E+02	3.104E+02	3.103E+02	3.103E+02	3.102E+02	3.101E+02
U-239	2.593E-04	2.587E-04	2.575E-04	2.568E-04	2.561E-04	2.548E-04	2.534E-04
U-240	1.087E-15	1.067E-15	1.028E-15	1.009E-15	9.904E-16	9.532E-16	9.176E-16
U-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-93	3.402E-01	3.416E-01	3.443E-01	3.456E-01	3.469E-01	3.496E-01	3.522E-01

TABLE XXI. Continued. (Pu-243 to Zr-93; 2.90% to 3.30%)

Nuclide	2.90%	3.00%	3.10%	3.15%	3.20%	3.25%	3.30%
Pu-243	1.713E-04	1.679E-04	1.645E-04	1.628E-04	1.611E-04	1.594E-04	1.576E-04
Pu-244	4.469E-05	4.296E-05	4.128E-05	4.047E-05	3.966E-05	3.888E-05	3.808E-05
Pu-245	4.131E-09	3.963E-09	3.800E-09	3.721E-09	3.643E-09	3.566E-09	3.488E-09
Pu-246	5.417E-11	5.185E-11	4.960E-11	4.850E-11	4.742E-11	4.637E-11	4.529E-11
Ru-106	9.501E-02	9.397E-02	9.291E-02	9.238E-02	9.185E-02	9.133E-02	9.078E-02
Sb-125	4.366E-03	4.335E-03	4.305E-03	4.289E-03	4.274E-03	4.259E-03	4.243E-03
Se-79	2.494E-03	2.503E-03	2.513E-03	2.518E-03	2.523E-03	2.528E-03	2.532E-03
Sm-151	6.580E-03	6.585E-03	6.591E-03	6.593E-03	6.597E-03	6.600E-03	6.603E-03
Sn-126	1.244E-02	1.233E-02	1.223E-02	1.218E-02	1.213E-02	1.208E-02	1.203E-02
Sr-90	2.441E-01	2.472E-01	2.503E-01	2.518E-01	2.534E-01	2.548E-01	2.564E-01
Tc-99	4.053E-01	4.064E-01	4.075E-01	4.080E-01	4.086E-01	4.091E-01	4.097E-01
Th-226	1.779E-18	1.781E-18	1.782E-18	1.781E-18	1.780E-18	1.779E-18	1.777E-18
Th-227	3.104E-15	3.116E-15	3.127E-15	3.131E-15	3.136E-15	3.139E-15	3.143E-15
Th-228	9.744E-10	9.777E-10	9.805E-10	9.816E-10	9.825E-10	9.833E-10	9.840E-10
Th-229	3.875E-10	3.862E-10	3.847E-10	3.838E-10	3.829E-10	3.819E-10	3.808E-10
Th-230	5.322E-09	5.362E-09	5.400E-09	5.418E-09	5.436E-09	5.453E-09	5.471E-09
Th-231	1.077E-11	1.104E-11	1.131E-11	1.144E-11	1.159E-11	1.173E-11	1.187E-11
Th-232	1.271E-07	1.307E-07	1.342E-07	1.359E-07	1.376E-07	1.393E-07	1.410E-07
Th-233	3.318E-13	3.395E-13	3.468E-13	3.504E-13	3.539E-13	3.572E-13	3.607E-13
Th-234	4.490E-09	4.488E-09	4.487E-09	4.486E-09	4.485E-09	4.485E-09	4.484E-09
U-230	1.730E-15	1.733E-15	1.733E-15	1.733E-15	1.732E-15	1.731E-15	1.729E-15
U-231	1.024E-13	1.026E-13	1.027E-13	1.027E-13	1.026E-13	1.026E-13	1.025E-13
U-232	1.474E-07	1.482E-07	1.489E-07	1.492E-07	1.494E-07	1.497E-07	1.499E-07
U-233	1.506E-07	1.553E-07	1.601E-07	1.625E-07	1.649E-07	1.673E-07	1.698E-07
U-234	1.514E-03	1.521E-03	1.528E-03	1.531E-03	1.534E-03	1.537E-03	1.539E-03
U-235	8.202E-01	8.792E-01	9.410E-01	9.734E-01	1.007E+00	1.040E+00	1.074E+00
U-236	1.325E+00	1.371E+00	1.418E+00	1.441E+00	1.464E+00	1.487E+00	1.510E+00
U-237	3.617E-03	3.704E-03	3.789E-03	3.830E-03	3.871E-03	3.911E-03	3.952E-03
U-238	3.100E+02	3.099E+02	3.098E+02	3.097E+02	3.097E+02	3.096E+02	3.096E+02
U-239	2.519E-04	2.504E-04	2.488E-04	2.480E-04	2.472E-04	2.463E-04	2.455E-04
U-240	8.827E-16	8.486E-16	8.154E-16	7.993E-16	7.834E-16	7.679E-16	7.520E-16
U-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zr-93	3.548E-01	3.574E-01	3.600E-01	3.613E-01	3.626E-01	3.638E-01	3.651E-01

TABLE XXI. Continued. (Pu-243 to Zr-93; 3.40% to 3.90%)

Nuclide	3.40%	3.45%	3.50%	3.60%	3.70%	3.80%	3.90%
Pu-243	1.542E-04	1.525E-04	1.508E-04	1.474E-04	1.440E-04	1.406E-04	1.372E-04
Pu-244	3.655E-05	3.581E-05	3.505E-05	3.363E-05	3.222E-05	3.088E-05	2.956E-05
Pu-245	3.340E-09	3.268E-09	3.194E-09	3.055E-09	2.918E-09	2.788E-09	2.659E-09
Pu-246	4.325E-11	4.226E-11	4.124E-11	3.934E-11	3.745E-11	3.566E-11	3.390E-11
Ru-106	8.972E-02	8.918E-02	8.862E-02	8.756E-02	8.646E-02	8.539E-02	8.429E-02
Sb-125	4.213E-03	4.198E-03	4.183E-03	4.153E-03	4.123E-03	4.093E-03	4.063E-03
Se-79	2.542E-03	2.546E-03	2.551E-03	2.560E-03	2.570E-03	2.578E-03	2.587E-03
Sm-151	6.610E-03	6.613E-03	6.617E-03	6.624E-03	6.632E-03	6.640E-03	6.649E-03
Sn-126	1.193E-02	1.188E-02	1.183E-02	1.173E-02	1.163E-02	1.154E-02	1.144E-02
Sr-90	2.593E-01	2.608E-01	2.623E-01	2.652E-01	2.681E-01	2.709E-01	2.737E-01
Tc-99	4.109E-01	4.114E-01	4.120E-01	4.131E-01	4.143E-01	4.154E-01	4.166E-01
Th-226	1.773E-18	1.770E-18	1.767E-18	1.759E-18	1.750E-18	1.740E-18	1.728E-18
Th-227	3.149E-15	3.152E-15	3.154E-15	3.158E-15	3.161E-15	3.163E-15	3.165E-15
Th-228	9.850E-10	9.852E-10	9.855E-10	9.855E-10	9.849E-10	9.842E-10	9.829E-10
Th-229	3.785E-10	3.773E-10	3.760E-10	3.734E-10	3.705E-10	3.675E-10	3.643E-10
Th-230	5.504E-09	5.520E-09	5.536E-09	5.566E-09	5.595E-09	5.624E-09	5.651E-09
Th-231	1.216E-11	1.231E-11	1.247E-11	1.278E-11	1.310E-11	1.343E-11	1.377E-11
Th-232	1.443E-07	1.460E-07	1.476E-07	1.508E-07	1.540E-07	1.571E-07	1.602E-07
Th-233	3.671E-13	3.702E-13	3.733E-13	3.790E-13	3.846E-13	3.898E-13	3.948E-13
Th-234	4.483E-09	4.482E-09	4.481E-09	4.479E-09	4.478E-09	4.476E-09	4.475E-09
U-230	1.725E-15	1.722E-15	1.719E-15	1.711E-15	1.703E-15	1.693E-15	1.682E-15
U-231	1.023E-13	1.022E-13	1.020E-13	1.016E-13	1.012E-13	1.006E-13	9.999E-14
U-232	1.502E-07	1.503E-07	1.505E-07	1.506E-07	1.507E-07	1.506E-07	1.505E-07
U-233	1.747E-07	1.772E-07	1.797E-07	1.847E-07	1.898E-07	1.948E-07	2.000E-07
U-234	1.544E-03	1.546E-03	1.547E-03	1.551E-03	1.553E-03	1.555E-03	1.557E-03
U-235	1.145E+00	1.182E+00	1.220E+00	1.296E+00	1.377E+00	1.460E+00	1.548E+00
U-236	1.556E+00	1.578E+00	1.601E+00	1.646E+00	1.691E+00	1.736E+00	1.781E+00
U-237	4.030E-03	4.067E-03	4.106E-03	4.178E-03	4.250E-03	4.317E-03	4.384E-03
U-238	3.095E+02	3.094E+02	3.094E+02	3.092E+02	3.091E+02	3.090E+02	3.089E+02
U-239	2.438E-04	2.429E-04	2.420E-04	2.402E-04	2.384E-04	2.365E-04	2.346E-04
U-240	7.219E-16	7.074E-16	6.924E-16	6.642E-16	6.364E-16	6.099E-16	5.838E-16
U-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zr-93	3.676E-01	3.688E-01	3.701E-01	3.725E-01	3.750E-01	3.773E-01	3.797E-01

TABLE XXI. Continued. (Pu-243 to Zr-93; 4.00% to 5.00%)

Nuclide	4.00%	4.10%	4.20%	4.40%	4.50%	4.60%	4.90%	5.00%
Pu-243	1.338E-04	1.305E-04	1.271E-04	1.206E-04	1.173E-04	1.141E-04	1.048E-04	1.017E-04
Pu-244	2.831E-05	2.709E-05	2.590E-05	2.366E-05	2.260E-05	2.158E-05	1.875E-05	1.787E-05
Pu-245	2.537E-09	2.420E-09	2.304E-09	2.087E-09	1.984E-09	1.886E-09	1.613E-09	1.529E-09
Pu-246	3.224E-11	3.063E-11	2.906E-11	2.611E-11	2.472E-11	2.341E-11	1.975E-11	1.863E-11
Ru-106	8.321E-02	8.213E-02	8.104E-02	7.887E-02	7.780E-02	7.674E-02	7.356E-02	7.251E-02
Sb-125	4.034E-03	4.005E-03	3.976E-03	3.919E-03	3.891E-03	3.863E-03	3.781E-03	3.754E-03
Se-79	2.596E-03	2.605E-03	2.613E-03	2.630E-03	2.639E-03	2.647E-03	2.670E-03	2.678E-03
Sm-151	6.658E-03	6.667E-03	6.678E-03	6.699E-03	6.710E-03	6.722E-03	6.759E-03	6.773E-03
Sn-126	1.135E-02	1.126E-02	1.117E-02	1.099E-02	1.091E-02	1.082E-02	1.057E-02	1.049E-02
Sr-90	2.765E-01	2.792E-01	2.819E-01	2.872E-01	2.898E-01	2.923E-01	2.997E-01	3.021E-01
Tc-99	4.177E-01	4.189E-01	4.201E-01	4.224E-01	4.236E-01	4.247E-01	4.282E-01	4.293E-01
Th-226	1.716E-18	1.702E-18	1.687E-18	1.654E-18	1.637E-18	1.618E-18	1.559E-18	1.539E-18
Th-227	3.166E-15	3.167E-15	3.168E-15	3.170E-15	3.171E-15	3.172E-15	3.178E-15	3.181E-15
Th-228	9.813E-10	9.794E-10	9.771E-10	9.718E-10	9.689E-10	9.656E-10	9.549E-10	9.509E-10
Th-229	3.610E-10	3.576E-10	3.541E-10	3.467E-10	3.429E-10	3.391E-10	3.272E-10	3.232E-10
Th-230	5.677E-09	5.703E-09	5.728E-09	5.776E-09	5.799E-09	5.821E-09	5.886E-09	5.908E-09
Th-231	1.412E-11	1.448E-11	1.486E-11	1.564E-11	1.604E-11	1.646E-11	1.779E-11	1.826E-11
Th-232	1.632E-07	1.661E-07	1.690E-07	1.747E-07	1.774E-07	1.801E-07	1.878E-07	1.903E-07
Th-233	3.994E-13	4.037E-13	4.077E-13	4.149E-13	4.181E-13	4.209E-13	4.279E-13	4.297E-13
Th-234	4.473E-09	4.471E-09	4.469E-09	4.466E-09	4.464E-09	4.462E-09	4.456E-09	4.454E-09
U-230	1.669E-15	1.656E-15	1.641E-15	1.610E-15	1.592E-15	1.575E-15	1.517E-15	1.497E-15
U-231	9.930E-14	9.855E-14	9.773E-14	9.592E-14	9.494E-14	9.393E-14	9.061E-14	8.943E-14
U-232	1.504E-07	1.501E-07	1.498E-07	1.489E-07	1.484E-07	1.479E-07	1.459E-07	1.452E-07
U-233	2.052E-07	2.104E-07	2.158E-07	2.265E-07	2.319E-07	2.373E-07	2.539E-07	2.594E-07
U-234	1.558E-03	1.558E-03	1.558E-03	1.557E-03	1.556E-03	1.555E-03	1.549E-03	1.547E-03
U-235	1.637E+00	1.730E+00	1.829E+00	2.032E+00	2.140E+00	2.250E+00	2.604E+00	2.728E+00
U-236	1.824E+00	1.868E+00	1.911E+00	1.997E+00	2.039E+00	2.081E+00	2.203E+00	2.243E+00
U-237	4.446E-03	4.506E-03	4.564E-03	4.671E-03	4.720E-03	4.766E-03	4.888E-03	4.923E-03
U-238	3.088E+02	3.086E+02	3.085E+02	3.082E+02	3.081E+02	3.080E+02	3.075E+02	3.074E+02
U-239	2.326E-04	2.306E-04	2.286E-04	2.244E-04	2.223E-04	2.202E-04	2.136E-04	2.113E-04
U-240	5.591E-16	5.351E-16	5.116E-16	4.674E-16	4.463E-16	4.263E-16	3.703E-16	3.529E-16
U-241	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zr-93	3.820E-01	3.843E-01	3.866E-01	3.911E-01	3.933E-01	3.955E-01	4.017E-01	4.038E-01

TABLE XXII. Summary of overall scores for each of the database nuclides, by PWR assembly class. (239Am to 251Cm)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Am-239	8.99E-01	9.06E-01	9.80E-01	9.66E-01	9.53E-01	9.54E-01	9.55E-01
Am-240	9.00E-01	9.08E-01	9.81E-01	9.66E-01	9.53E-01	9.56E-01	9.54E-01
Am-241	9.64E-01	9.70E-01	9.92E-01	9.68E-01	9.26E-01	9.80E-01	9.17E-01
Am-242	8.99E-01	9.06E-01	9.80E-01	9.66E-01	9.53E-01	9.43E-01	9.55E-01
Am-242m	9.57E-01	9.65E-01	9.93E-01	9.67E-01	9.28E-01	9.79E-01	9.18E-01
Am-243	7.18E-01	7.23E-01	8.93E-01	9.09E-01	7.55E-01	7.73E-01	9.08E-01
Am-244	6.56E-01	6.63E-01	8.65E-01	9.08E-01	6.60E-01	7.27E-01	9.06E-01
Am-245	5.44E-01	5.46E-01	7.97E-01	8.78E-01	5.01E-01	6.18E-01	8.76E-01
Am-246	4.93E-01	4.90E-01	7.69E-01	8.75E-01	4.22E-01	5.72E-01	8.70E-01
Bk-249	2.88E-01	2.18E-01	5.84E-01	7.18E-01	1.71E-01	3.51E-01	7.41E-01
Bk-250	2.68E-01	1.89E-01	5.60E-01	7.16E-01	1.31E-01	3.31E-01	7.50E-01
Bk-251	2.49E-01	1.58E-01	5.26E-01	7.08E-01	9.28E-02	3.06E-01	7.55E-01
Cd-113m	9.25E-01	9.27E-01	9.61E-01	9.56E-01	9.52E-01	9.42E-01	9.49E-01
Cf-249	2.87E-01	2.15E-01	5.81E-01	7.02E-01	1.80E-01	3.47E-01	7.23E-01
Cf-250	2.77E-01	2.00E-01	5.70E-01	7.08E-01	1.54E-01	3.36E-01	7.35E-01
Cf-251	2.67E-01	1.84E-01	5.58E-01	7.00E-01	1.38E-01	3.28E-01	7.30E-01
Cf-252	2.38E-01	1.34E-01	5.04E-01	6.65E-01	8.65E-02	2.85E-01	7.11E-01
Cf-253	2.25E-01	1.13E-01	4.78E-01	6.59E-01	6.28E-02	2.71E-01	7.16E-01
Cf-254	2.10E-01	8.81E-02	4.42E-01	6.42E-01	4.02E-02	2.50E-01	7.15E-01
Cf-255	2.01E-01	7.55E-02	4.21E-01	6.40E-01	2.89E-02	2.41E-01	7.29E-01
Cm-241	8.08E-01	8.18E-01	9.43E-01	9.52E-01	8.41E-01	8.78E-01	9.53E-01
Cm-242	8.67E-01	8.73E-01	9.67E-01	9.55E-01	9.23E-01	9.15E-01	9.49E-01
Cm-243	7.91E-01	8.00E-01	9.38E-01	9.37E-01	8.44E-01	8.58E-01	9.39E-01
Cm-244	5.90E-01	5.92E-01	8.23E-01	8.67E-01	6.00E-01	6.57E-01	8.63E-01
Cm-245	5.34E-01	5.35E-01	7.95E-01	8.49E-01	5.34E-01	6.20E-01	8.47E-01
Cm-246	4.32E-01	4.16E-01	7.23E-01	8.04E-01	3.93E-01	5.03E-01	8.05E-01
Cm-247	3.69E-01	3.35E-01	6.72E-01	7.72E-01	3.02E-01	4.40E-01	7.79E-01
Cm-248	3.11E-01	2.54E-01	6.10E-01	7.33E-01	2.10E-01	3.74E-01	7.50E-01
Cm-249	2.89E-01	2.21E-01	5.85E-01	7.32E-01	1.63E-01	3.55E-01	7.58E-01
Cm-250	2.53E-01	1.61E-01	5.27E-01	6.92E-01	1.03E-01	3.08E-01	7.33E-01
Cm-251	2.38E-01	1.38E-01	5.04E-01	6.91E-01	7.69E-02	2.91E-01	7.44E-01

TABLE XXII. Continued. (^{134}Cs to ^{242}Pu)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Cs-134	9.40E-01	9.43E-01	9.78E-01	9.94E-01	9.26E-01	9.53E-01	9.74E-01
Cs-135	9.39E-01	9.41E-01	9.74E-01	9.82E-01	9.36E-01	9.37E-01	9.80E-01
Cs-137	9.99E-01	9.99E-01	9.99E-01	9.99E-01	9.99E-01	9.99E-01	9.99E-01
Eu-154	9.36E-01	9.41E-01	9.82E-01	9.83E-01	9.50E-01	9.62E-01	9.83E-01
Eu-155	9.38E-01	9.42E-01	9.82E-01	9.82E-01	9.53E-01	9.53E-01	9.82E-01
He-4	8.98E-01	9.06E-01	9.55E-01	9.58E-01	9.15E-01	9.14E-01	9.56E-01
I-129	9.50E-01	9.52E-01	9.79E-01	9.78E-01	9.64E-01	9.56E-01	9.77E-01
Kr-85	9.28E-01	9.31E-01	9.64E-01	9.66E-01	9.38E-01	9.31E-01	9.64E-01
Nb-94	8.70E-01	8.73E-01	9.48E-01	9.46E-01	9.06E-01	8.87E-01	9.45E-01
Np-235	9.70E-01	9.59E-01	9.96E-01	9.70E-01	9.27E-01	9.83E-01	9.30E-01
Np-236	9.41E-01	9.50E-01	9.53E-01	9.51E-01	9.52E-01	9.11E-01	9.41E-01
Np-236m	9.76E-01	9.87E-01	9.71E-01	9.48E-01	9.24E-01	9.47E-01	9.10E-01
Np-237	9.21E-01	9.31E-01	9.46E-01	9.46E-01	9.32E-01	9.09E-01	9.40E-01
Np-238	9.76E-01	9.86E-01	9.71E-01	9.48E-01	9.25E-01	9.55E-01	9.11E-01
Np-239	9.29E-01	9.29E-01	9.76E-01	9.77E-01	9.47E-01	9.41E-01	9.78E-01
Np-240	8.60E-01	8.65E-01	9.48E-01	9.75E-01	8.47E-01	8.85E-01	9.58E-01
Np-240m	5.97E-01	6.03E-01	8.25E-01	8.79E-01	5.88E-01	6.57E-01	8.75E-01
Pa-231	7.44E-01	7.35E-01	8.50E-01	8.82E-01	7.26E-01	7.43E-01	8.70E-01
Pa-232	7.77E-01	7.69E-01	8.70E-01	8.83E-01	7.84E-01	7.74E-01	8.80E-01
Pa-233	9.23E-01	9.32E-01	9.47E-01	9.46E-01	9.35E-01	9.10E-01	9.40E-01
Pa-234	9.78E-01	9.89E-01	9.73E-01	9.50E-01	9.27E-01	9.52E-01	9.13E-01
Pa-234m	9.95E-01	9.93E-01	9.95E-01	9.81E-01	9.50E-01	9.86E-01	9.51E-01
Pd-107	8.55E-01	8.59E-01	9.40E-01	9.42E-01	8.86E-01	8.73E-01	9.41E-01
Pm-147	9.39E-01	9.40E-01	9.69E-01	9.87E-01	9.21E-01	9.42E-01	9.69E-01
Pu-236	9.80E-01	9.70E-01	9.89E-01	9.71E-01	9.43E-01	9.76E-01	9.38E-01
Pu-237	8.99E-01	9.02E-01	9.80E-01	9.89E-01	8.72E-01	9.59E-01	9.38E-01
Pu-238	9.56E-01	9.55E-01	9.97E-01	9.84E-01	9.49E-01	9.94E-01	9.58E-01
Pu-239	9.91E-01	9.91E-01	9.96E-01	9.79E-01	9.47E-01	9.84E-01	9.47E-01
Pu-240	9.68E-01	9.68E-01	9.97E-01	9.75E-01	9.54E-01	9.72E-01	9.48E-01
Pu-241	9.39E-01	9.43E-01	9.89E-01	9.70E-01	9.58E-01	9.69E-01	9.50E-01
Pu-242	8.24E-01	8.27E-01	9.40E-01	9.39E-01	8.69E-01	8.54E-01	9.40E-01

TABLE XXII. Continued. (^{243}Pu to ^{93}Zr)

Nuclide	BW15x15	CE14x14	CE16x16	CE16x16 Sys80	W14x14	W15x15	W17x17
Pu-243	7.57E-01	7.64E-01	9.12E-01	9.37E-01	7.70E-01	8.12E-01	9.36E-01
Pu-244	5.97E-01	6.02E-01	8.25E-01	8.79E-01	5.88E-01	6.57E-01	8.75E-01
Pu-245	5.44E-01	5.46E-01	7.97E-01	8.78E-01	5.01E-01	6.18E-01	8.76E-01
Pu-246	4.93E-01	4.90E-01	7.69E-01	8.74E-01	4.21E-01	5.72E-01	8.70E-01
Ru-106	8.89E-01	8.91E-01	9.57E-01	9.61E-01	9.09E-01	9.04E-01	9.60E-01
Sb-125	9.39E-01	9.40E-01	9.74E-01	9.75E-01	9.52E-01	9.45E-01	9.74E-01
Se-79	9.73E-01	9.74E-01	9.86E-01	9.88E-01	9.74E-01	9.74E-01	9.87E-01
Sm-151	9.89E-01	9.91E-01	9.95E-01	9.95E-01	9.76E-01	9.78E-01	9.82E-01
Sn-126	9.31E-01	9.34E-01	9.70E-01	9.71E-01	9.46E-01	9.38E-01	9.69E-01
Sr-90	9.22E-01	9.26E-01	9.60E-01	9.63E-01	9.32E-01	9.25E-01	9.60E-01
Tc-99	9.79E-01	9.80E-01	9.88E-01	9.94E-01	9.73E-01	9.79E-01	9.92E-01
Th-226	9.03E-01	9.05E-01	9.71E-01	9.91E-01	8.60E-01	9.54E-01	9.32E-01
Th-227	9.94E-01	9.84E-01	9.86E-01	9.91E-01	9.38E-01	9.84E-01	9.56E-01
Th-228	9.75E-01	9.73E-01	9.99E-01	9.97E-01	9.75E-01	9.94E-01	9.85E-01
Th-229	9.03E-01	9.06E-01	9.67E-01	9.89E-01	8.86E-01	9.45E-01	9.64E-01
Th-230	9.61E-01	9.67E-01	9.81E-01	9.82E-01	9.65E-01	9.49E-01	9.78E-01
Th-231	8.25E-01	8.28E-01	9.40E-01	9.38E-01	8.61E-01	8.44E-01	9.39E-01
Th-232	8.66E-01	8.74E-01	9.22E-01	9.29E-01	8.76E-01	8.65E-01	9.22E-01
Th-233	9.13E-01	9.26E-01	9.47E-01	9.30E-01	9.20E-01	9.01E-01	9.03E-01
Th-234	9.94E-01	9.92E-01	9.96E-01	9.78E-01	9.46E-01	9.84E-01	9.46E-01
U-230	9.03E-01	9.05E-01	9.71E-01	9.91E-01	8.60E-01	9.54E-01	9.32E-01
U-231	9.07E-01	9.08E-01	9.73E-01	9.89E-01	8.64E-01	9.57E-01	9.32E-01
U-232	9.75E-01	9.72E-01	9.99E-01	9.88E-01	9.63E-01	9.88E-01	9.66E-01
U-233	8.30E-01	8.32E-01	9.05E-01	9.23E-01	8.24E-01	8.22E-01	9.14E-01
U-234	9.97E-01	9.92E-01	9.93E-01	9.88E-01	9.81E-01	9.78E-01	9.76E-01
U-235	7.20E-01	7.07E-01	8.05E-01	8.52E-01	6.83E-01	7.08E-01	8.30E-01
U-236	8.36E-01	8.41E-01	9.01E-01	9.12E-01	8.42E-01	8.32E-01	9.05E-01
U-237	9.00E-01	9.09E-01	9.36E-01	9.26E-01	9.22E-01	8.90E-01	9.09E-01
U-238	9.94E-01	9.92E-01	9.96E-01	9.78E-01	9.46E-01	9.84E-01	9.46E-01
U-239	9.29E-01	9.28E-01	9.76E-01	9.77E-01	9.47E-01	9.41E-01	9.78E-01
U-240	5.97E-01	6.03E-01	8.25E-01	8.79E-01	5.88E-01	6.57E-01	8.75E-01
Zr-93	9.51E-01	9.53E-01	9.75E-01	9.78E-01	9.57E-01	9.53E-01	9.76E-01

TABLE XXIII. Summary of overall scores for each of the database nuclides, by BWR and individual assembly classes. (239Am to 249Cm)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Am-239	5.52E-01	7.57E-01	8.48E-01	8.43E-01	9.27E-01	9.40E-01
Am-240	5.52E-01	7.62E-01	8.52E-01	8.45E-01	9.29E-01	9.41E-01
Am-241	7.42E-01	9.40E-01	9.15E-01	9.80E-01	9.85E-01	9.93E-01
Am-242	5.55E-01	7.59E-01	8.49E-01	8.44E-01	9.28E-01	9.40E-01
Am-242m	7.10E-01	9.30E-01	9.35E-01	9.70E-01	9.81E-01	9.89E-01
Am-243	4.08E-01	3.55E-01	4.85E-01	5.50E-01	7.77E-01	7.90E-01
Am-244	5.51E-01	2.36E-01	3.10E-01	4.33E-01	7.21E-01	7.39E-01
Am-245	8.71E-01	1.17E-01	1.43E-01	2.63E-01	6.18E-01	6.33E-01
Am-246	6.48E-01	7.00E-02	7.99E-02	1.84E-01	5.64E-01	5.83E-01
Bk-249	5.87E-01	1.29E-02	1.58E-02	2.09E-02	3.09E-01	3.15E-01
Bk-250	4.69E-01	8.18E-03	7.70E-03	1.04E-02	2.72E-01	2.81E-01
Bk-251	4.15E-01	5.43E-03	3.12E-03	4.27E-03	2.31E-01	2.41E-01
Cd-113m	8.89E-01	8.92E-01	9.63E-01	9.16E-01	9.46E-01	9.40E-01
Cf-249	7.40E-01	1.51E-02	1.93E-02	2.26E-02	3.09E-01	3.12E-01
Cf-250	5.39E-01	1.05E-02	1.22E-02	1.53E-02	2.88E-01	2.95E-01
Cf-251	5.37E-01	9.18E-03	9.35E-03	1.11E-02	2.71E-01	2.78E-01
Cf-252	4.56E-01	5.82E-03	3.63E-03	3.22E-03	2.09E-01	2.14E-01
Cf-253	4.16E-01	4.32E-03	1.42E-03	1.20E-03	1.77E-01	1.84E-01
Cf-254	3.93E-01	3.21E-03	4.05E-04	2.85E-04	1.40E-01	1.47E-01
Cf-255	3.81E-01	2.50E-03	1.02E-04	7.49E-05	1.17E-01	1.25E-01
Cm-241	4.46E-01	5.33E-01	5.97E-01	6.73E-01	8.52E-01	8.68E-01
Cm-242	5.12E-01	6.82E-01	7.95E-01	7.93E-01	9.01E-01	9.14E-01
Cm-243	3.99E-01	5.06E-01	6.10E-01	6.58E-01	8.40E-01	8.56E-01
Cm-244	4.57E-01	1.81E-01	2.63E-01	3.54E-01	6.65E-01	6.76E-01
Cm-245	4.60E-01	1.34E-01	1.83E-01	2.72E-01	6.16E-01	6.26E-01
Cm-246	7.51E-01	6.50E-02	8.77E-02	1.45E-01	5.09E-01	5.16E-01
Cm-247	8.06E-01	3.60E-02	4.91E-02	8.03E-02	4.31E-01	4.38E-01
Cm-248	6.60E-01	1.86E-02	2.50E-02	3.57E-02	3.49E-01	3.54E-01
Cm-249	5.13E-01	1.13E-02	1.31E-02	1.93E-02	3.09E-01	3.17E-01

TABLE XXIII. Continued. (^{250}Cm to ^{241}Pu)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Cm-250	4.58E-01	6.77E-03	5.58E-03	6.29E-03	2.40E-01	2.45E-01
Cm-251	4.10E-01	4.76E-03	2.26E-03	2.65E-03	2.08E-01	2.16E-01
Cs-134	8.59E-01	8.23E-01	8.15E-01	8.84E-01	9.51E-01	9.56E-01
Cs-135	8.61E-01	8.61E-01	8.72E-01	9.01E-01	9.51E-01	9.52E-01
Cs-137	9.97E-01	9.97E-01	9.98E-01	9.98E-01	9.99E-01	9.98E-01
Eu-154	7.91E-01	8.37E-01	8.60E-01	8.89E-01	9.52E-01	9.58E-01
Eu-155	8.11E-01	8.44E-01	8.75E-01	8.95E-01	9.53E-01	9.58E-01
He-4	8.28E-01	8.28E-01	8.49E-01	8.47E-01	9.27E-01	9.23E-01
I-129	8.93E-01	9.03E-01	9.40E-01	9.29E-01	9.64E-01	9.62E-01
Kr-85	8.88E-01	8.79E-01	9.08E-01	8.98E-01	9.47E-01	9.41E-01
Nb-94	6.79E-01	7.17E-01	8.34E-01	8.14E-01	9.01E-01	9.03E-01
Np-235	8.39E-01	7.71E-01	7.64E-01	9.02E-01	9.60E-01	9.85E-01
Np-236	9.20E-01	9.62E-01	9.67E-01	9.44E-01	9.66E-01	9.43E-01
Np-236m	9.04E-01	8.51E-01	8.08E-01	9.43E-01	9.96E-01	9.76E-01
Np-237	9.72E-01	9.43E-01	9.53E-01	9.06E-01	9.50E-01	9.29E-01
Np-238	9.13E-01	8.54E-01	8.10E-01	9.44E-01	9.96E-01	9.75E-01
Np-239	7.83E-01	8.08E-01	8.94E-01	8.94E-01	9.43E-01	9.49E-01
Np-240	7.56E-01	6.16E-01	6.58E-01	7.54E-01	8.86E-01	8.97E-01
Np-240m	6.31E-01	1.88E-01	2.47E-01	3.57E-01	6.71E-01	6.82E-01
Pa-231	5.79E-01	5.77E-01	6.14E-01	6.36E-01	7.74E-01	7.66E-01
Pa-232	6.81E-01	6.36E-01	7.02E-01	6.91E-01	8.09E-01	7.96E-01
Pa-233	9.72E-01	9.46E-01	9.55E-01	9.09E-01	9.51E-01	9.30E-01
Pa-234	9.23E-01	8.66E-01	8.25E-01	9.47E-01	9.94E-01	9.79E-01
Pa-234m	9.58E-01	9.74E-01	8.83E-01	9.68E-01	1.00E+00	9.98E-01
Pd-107	6.71E-01	6.89E-01	7.96E-01	7.88E-01	8.90E-01	8.90E-01
Pm-147	9.04E-01	8.69E-01	8.60E-01	8.97E-01	9.49E-01	9.48E-01
Pu-236	8.12E-01	8.10E-01	8.03E-01	9.22E-01	9.70E-01	9.95E-01
Pu-237	7.05E-01	6.52E-01	6.16E-01	7.78E-01	9.11E-01	9.36E-01
Pu-238	7.44E-01	7.96E-01	8.00E-01	8.92E-01	9.58E-01	9.79E-01
Pu-239	9.49E-01	9.72E-01	8.88E-01	9.71E-01	9.99E-01	1.00E+00
Pu-240	8.26E-01	9.31E-01	9.43E-01	9.84E-01	9.80E-01	9.85E-01
Pu-241	7.14E-01	8.67E-01	9.74E-01	9.27E-01	9.58E-01	9.67E-01

TABLE XXIII. Continued. (^{242}Pu to ^{93}Zr)

Nuclide	GE BWR 2,3	GE BWR 4,6	Fort Calhoun	Palisades	St. Lucie unit 2	South Texas
Pu-242	5.06E-01	5.70E-01	7.09E-01	7.25E-01	8.63E-01	8.74E-01
Pu-243	5.13E-01	4.07E-01	4.92E-01	5.93E-01	8.07E-01	8.23E-01
Pu-244	6.31E-01	1.88E-01	2.47E-01	3.57E-01	6.71E-01	6.82E-01
Pu-245	8.71E-01	1.17E-01	1.43E-01	2.63E-01	6.18E-01	6.33E-01
Pu-246	6.48E-01	7.00E-02	7.99E-02	1.84E-01	5.64E-01	5.83E-01
Ru-106	7.28E-01	7.41E-01	8.27E-01	8.32E-01	9.13E-01	9.16E-01
Sb-125	8.64E-01	8.70E-01	9.18E-01	9.11E-01	9.54E-01	9.53E-01
Se-79	9.60E-01	9.50E-01	9.59E-01	9.60E-01	9.80E-01	9.78E-01
Sm-151	9.48E-01	9.60E-01	9.35E-01	9.70E-01	9.93E-01	9.92E-01
Sn-126	8.56E-01	8.60E-01	9.09E-01	9.01E-01	9.49E-01	9.47E-01
Sr-90	8.81E-01	8.71E-01	9.00E-01	8.90E-01	9.42E-01	9.36E-01
Tc-99	9.75E-01	9.61E-01	9.56E-01	9.66E-01	9.83E-01	9.82E-01
Th-226	8.57E-01	7.10E-01	6.25E-01	7.92E-01	9.13E-01	9.30E-01
Th-227	6.61E-01	8.49E-01	9.58E-01	9.84E-01	9.87E-01	9.79E-01
Th-228	9.33E-01	9.45E-01	8.99E-01	9.47E-01	9.76E-01	9.85E-01
Th-229	8.21E-01	7.69E-01	7.09E-01	8.17E-01	9.19E-01	9.28E-01
Th-230	9.44E-01	9.56E-01	9.57E-01	9.45E-01	9.75E-01	9.69E-01
Th-231	6.03E-01	6.43E-01	7.08E-01	7.30E-01	8.59E-01	8.67E-01
Th-232	8.25E-01	8.11E-01	8.31E-01	8.15E-01	9.01E-01	8.85E-01
Th-233	9.30E-01	9.23E-01	9.51E-01	9.20E-01	9.51E-01	9.28E-01
Th-234	9.58E-01	9.73E-01	8.78E-01	9.67E-01	9.99E-01	9.98E-01
U-230	8.57E-01	7.10E-01	6.24E-01	7.91E-01	9.12E-01	9.31E-01
U-231	8.57E-01	7.14E-01	6.27E-01	7.96E-01	9.15E-01	9.33E-01
U-232	9.17E-01	9.13E-01	8.61E-01	9.36E-01	9.72E-01	9.86E-01
U-233	7.44E-01	7.19E-01	7.49E-01	7.56E-01	8.62E-01	8.50E-01
U-234	9.54E-01	9.41E-01	9.41E-01	9.82E-01	9.92E-01	9.98E-01
U-235	5.70E-01	5.60E-01	5.91E-01	6.11E-01	7.48E-01	7.31E-01
U-236	7.84E-01	7.60E-01	7.88E-01	7.73E-01	8.73E-01	8.55E-01
U-237	9.39E-01	9.12E-01	9.46E-01	8.92E-01	9.35E-01	9.12E-01
U-238	9.57E-01	9.73E-01	8.79E-01	9.67E-01	9.99E-01	9.98E-01
U-239	7.83E-01	8.08E-01	8.94E-01	8.93E-01	9.43E-01	9.49E-01
U-240	6.31E-01	1.88E-01	2.47E-01	3.57E-01	6.71E-01	6.82E-01
Zr-93	9.23E-01	9.15E-01	9.34E-01	9.30E-01	9.64E-01	9.60E-01